November 23, 1999

Mr. J. N. Adkins
Vice President - Production
United States Enrichment Corporation
Two Democracy Center
6903 Rockledge Drive
Bethesda, MD 20817

SUBJECT: PADUCAH AND PORTSMOUTH SPECIAL INSPECTION REPORT 70-7001 AND 70-7002/99013(DNMS)

Dear Mr. Adkins:

On October 28, 1999, the NRC completed a special team inspection of your radiation protection programs at the Paducah and Portsmouth Gaseous Diffusion Plants. The purpose of the inspection was to determine whether activities authorized by the certificates were conducted safely. At the conclusion of the inspection, the team leader discussed the findings with you and members of your staff at exit meetings held on October 27, and 28, 1999. These meetings were open to public observation.

The enclosed copy of our Special Safety Team report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

The Special Safety Team Inspection was conducted in response to concerns over past worker exposures to transuranic and fission product radionuclides at the Paducah Gaseous Diffusion Plant (PGDP). The team inspected the current radiological work practices at both the PGDP and Portsmouth (PORTS) gaseous diffusion plants. The purpose of the inspections was to confirm that the radiation protection and control programs at both sites were adequate to protect the workers and the general public. The inspections were also intended to ensure that the programs were being operated in a manner consistent with their respective applications for certification.

The team concluded that, overall, United States Enrichment Corporation has established good radiological control programs at both PGDP and PORTS, that were adequate to protect the workers and the public. In addition, the team concluded good controls existed over site releases, and your environmental monitoring programs were of high quality and adequately assessed the impact of operations on the environment. Several weaknesses were identified which were discussed at the exit meetings and with your staff during the inspections. The first weakness observed appears to be generic to both sites and involves the adequacy of site-specific training. Our review of training material, direct field observations, and interviews of workers, identified that site specific information regarding radiological hazards from transuranics was not included in the material, and some workers were not aware of site radiological hazards or requirements imposed due to the radiological environment. The second weakness involved the assumptions made in your calculations for internal dose assessment at PGDP. The team agreed that the general site airborne activity level at PGDP is low; however, the assumption of zero contribution to internal dose from sources other than uranium was not adequately supported by recent data.

It was not the responsibility of this Special Safety Team to determine compliance with NRC rules and regulations or to recommend enforcement actions. These aspects will be reviewed in a subsequent inspection. In addition to the two specific concerns discussed above, several Inspection Follow-up Issues were identified during the course of the inspections. These issues will be reviewed in a subsequent inspection. We request that you provide a written response within 30 days of the date of this letter addressing your review and any actions you plan to take to address the two weaknesses and any additional information you can provide on the Inspection Follow-up Issues.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure and your response will be placed in the NRC Public Document Room.

We will gladly discuss any questions you have concerning these observations.

Sincerely,

/s/ C. D. Pederson

Cynthia D. Pederson, Director
Division of Nuclear Materials Safety

Docket Nos. 70-7001 and 70-7002
Certificate Nos. GDP-1 and GDP-2

Enclosure: Inspection Report 70-7001 & 70-7002/99013(DNMS)

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REGION III

Docket Nos:	70-7001 70-7002
Certificate Nos:	GDP-1 GDP-2
Report No:	70-7001 and 7002/99013(DNMS)
Facility Operator:	United States Enrichment Corporation
Facility Name:	Paducah Gaseous Diffusion Plant Portsmouth Gaseous Diffusion Plant
Locations:	Paducah Gaseous Diffusion Plant 5600 Hobbs Road Paducah, KY 42001 Portsmouth Gaseous Diffusion Plant 3930 U.S. Route 23 South Piketon, OH 45661
Dates:	August 30 through October 28, 1999
Inspectors:	Patrick L. Hiland, Team Leader Sami Sherbini, Senior Health Physicist, NMSS/IMNS William Snell, Health Physics Manager, Region III James Kottan, Health Physics Manager, Region I Yawar Faraz, Project Manager, CHP, NMSS
Approved By:	Cynthia D. Pederson, Director Division of Nuclear Materials Safety

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OVERVIEW

As a result of concerns over past worker exposures to transuranic and fission product radionuclides at the Paducah Gaseous Diffusion Plant (PGDP), the NRC conducted a special inspection of the current radiological work practices at the USEC leased portions of both PGDP and the Portsmouth (PORTS) gaseous diffusion plants (GDP). The purpose of the inspections was to ensure that the radiation protection and control programs at both sites were adequate to protect the workers and the general public. The inspections were also intended to ensure that the programs were being operated in a manner consistent with their respective applications for certification.

Areas reviewed during this inspection included radiological site characterization, organization, qualifications and training of the health physics staff, radiation protection controls, the facilities' in-plant air sampling programs, external and internal dose assessment programs, and the environmental and effluent monitoring programs.

BACKGROUND

Intermittently between 1953 and 1975, recycled uranium was used at PGDP to manufacture uranium hexafluoride (UF_6), which is the feed material used in the gaseous diffusion process. Recycled uranium (RU) was extracted from spent nuclear reactor fuel during processing of that fuel at chemical reprocessing plants. Reprocessing did not remove all impurities from the uranium, and some radionuclides, mainly technetium and the transuranic elements plutonium and neptunium, remained in the RU as trace impurities. These trace impurities were then introduced into the site environs as contaminants both during the manufacture of UF_6 and also during the diffusion process. Most of the plutonium and neptunium that was fed into the cascade was believed to have been deposited on the inside surfaces of the process equipment very close to the feed points, but some neptunium was believed to have also been deposited on the inside surfaces of the process equipment in all areas of the cascade. Estimates of the amounts of transuranics and other non-uranium radionuclides that may have been fed into the cascade have been made in the past. For example, a March 1984 Union Carbide report estimated that until that time, a total of 539 kilograms (9 Ci) of technetium (^{99}Tc), 4.6 kilograms (3.3 Ci) of neptunium (^{237}Np), and 0.1 grams (6.3 mCi) of plutonium (^{239}Pu) were fed into the cascade. Later, a September 1990 Oak Ridge report indicated that about 2.3 kilograms of ^{237}Np may have been fed to the cascade of which 1.3 kilograms was removed during the two cascade improvement projects of 1958-1962 and 1974-1981. During the cascade improvement programs, equipment was removed from the cascade and decontaminated in Building C-400, which houses the site's decontamination facility, and about 75 percent of the enrichment barriers (providing 99 percent of the internal surface area of the cascade) were replaced by new enrichment barriers. The old enrichment barriers were decontaminated and smelted forming nickel ingots which were being stored in Department of Energy (DOE) areas at PGDP. The amount of plutonium believed to have been fed into the cascade was small since most of the plutonium tended to remain in the feed cylinder heels and would be removed during periodic cleaning of the feed cylinders (ref. Ky-815 "The TRU Story").

At PORTS, RU was also converted from an oxide form to UF_6 in the northeast corner of Building X-705 (currently sealed and under DOE regulatory oversight) to be re-fed into the cascade. However, unlike PGDP, the RU handled at PORTS consisted of highly enriched uranium (^{235}U concentrations up to almost 100 percent by weight). This operation ended in 1975. No estimates were available regarding the amounts of transuranics and other non-uranium radionuclides that may have been fed into the PORTS cascade. However, the NRC staff believes the PORTS quantities are lower than the quantities that may have been fed into the PGDP cascade. At the time of this inspection, the DOE was conducting an historical review of this operation. The results of DOE's review may include estimates of quantities of non-uranium radionuclides that may have been fed into the PORTS cascade.

EXTERNAL AND INTERNAL RADIATION HAZARDS

External radiation levels at both sites were low, and annual exposures to workers from external sources were below 10 percent of the regulatory limit, with most monitored workers receiving no measurable dose from external radiation. Internal radiation hazards at both sites varied with location, partly because of differences in the nature of work at these locations, but also because of variations in the fractional levels of transuranic radionuclides that co-exited with uranium contaminations at these locations. However, as indicated by the results of the bioassay programs at both sites, exposures to airborne radioactive materials were quite low in all locations, with most monitored workers showing no measurable internal doses. The internal doses to workers that did show intakes were estimated by the certificatee to be quite small.

The data reviewed by the special inspection team indicated that the process streams at both PGDP and PORTS contained only trace amounts of transuranic materials, but there still remained legacy transuranic materials in the form of contamination inside equipment and on various equipment and building surfaces. The data also suggested that the levels of the transuranics, although falling over time at both sites, were not yet negligible in some areas and, if inhaled, could deliver internal doses that are substantially larger than the doses received from uranium.

The United States Enrichment Corporation's (USEC) application for certification assumed transuranics to constitute a maximum of 8 percent of the total radiological activity in USEC's leased areas at PGDP, and 2 percent or 0.5 percent at PORTS, depending on location. At PORTS isotopic analyses of air and removable surface contamination samples obtained between 1993 and 1998 from areas that were currently leased by USEC indicated that the assumption of two percent as the maximum transuranic activity concentration was conservative. At PGDP, sufficient isotopic data was not provided at the time of the special safety inspection to make a reliable determination of the appropriate level of transuranic activity, but it was clear from the available data that 8 percent was very conservative for most areas of the plant.

During the internal dose assessment process at PORTS, either the 2 percent value or the 0.5 percent value for radiological activity due to transuranics was being used. However, at PGDP a value of zero was being used to assess internal doses even though the certificatee's application considers transuranics to exist at levels as high as 8 percent. In addition, the validity of a number of assumptions, some not necessarily conservative, that were being used at PGDP in assessing internal doses had not been adequately established. These assumptions could, under certain circumstances, result in underestimating internal doses. However, the inspection team concluded that conditions at the PGDP are such that, even though internal doses may have been underestimated in some cases as a result of using non-conservative assumptions, the actual internal doses received are not likely to have exceeded any regulatory limits.

RADIOLOGICAL CONTROLS AND PROTECTION MEASURES

The in-plant air monitoring programs at both PORTS and PGDP were found to be extensive and well run. Large numbers of samples and measurements were routinely made with state of the art equipment in most cases. The health physics staff was found to be well qualified and competent, and management appeared to support radiation controls initiatives and routine activities. Training of the health physics staff was also found to be strong. The plant staff's understanding of required emergency and protective actions such as "see-and-flee" was adequate. However, plant staff (operations, maintenance, etc.) at both sites demonstrated a weak understanding of the significance of radiological and chemical hazards that could be encountered during normal operations.

Procedures were found to be generally adequate, but many procedures, as well as technical basis documents, were out-of-date at both sites. Because of this, some program elements at both sites were not clearly defined, and practices often differed from written procedures and technical basis documents in some program areas. In some cases, actual practices were more appropriate than the practices described in the written documents, but in others, the bases for current practices were unclear. Occasionally, site staff did not agree on actual requirements, which were known only by those closely associated with the activity in question. At the time of this inspection, the staff at both sites were aware of these deficiencies and were working to correct the problems.

The radiation worker training was generally good; however, the training material was sometimes outdated and sometimes unclear. It was also found that there was little site specific information on the radiological environment at the plants, particularly with respect to transuranic contamination. The current training material was neither clear or easy to understand, and as a result did not clearly describe and explain the radiological environment at the site.

Radiological controls were found to be generally effective with some weaknesses observed by the team. These included items staged across radiological boundaries, poor housekeeping, improper personnel exit monitoring and unjustified criteria for determining contaminated areas at PGDP.

ENVIRONMENTAL AND EFFLUENT MONITORING

The airborne radioactive effluent control programs were determined to be effective for the control of airborne radioactive emissions from the USEC facilities. Additionally, the overall control of liquid effluent releases and the water sampling programs were determined to be of high quality and adequate to collect representative water samples for the analysis of radioactive releases from the sites. The cognizant USEC staffs were knowledgeable and experienced in the area of effluent monitoring, and the airborne radioactive effluent control program functioned as outlined in Section 5.1 of the Safety Analysis Reports (SARs). Offsite doses to members of the public were a small fraction of the regulatory limits.

The radiological environmental monitoring programs (REMPs) conducted at the GDPs were observed to be of high quality and adequately assessed the impact of the GDPs' operations on the environment in the immediate vicinity of the plant sites. The REMP was effective in supplementing the radiological effluent monitoring program by verifying anticipated concentrations of radioactivity in the environment and related exposures to members of the public. The ambient air monitoring stations were observed to be of high quality and were well maintained. The soil, sediment and vegetation sampling programs were adequate to monitor the environmental impact of the GDP's emissions on the environs surrounding the sites. The direct radiation monitoring programs being conducted to monitor external gamma radiation were also adequate.

Although some inconsistencies with industry practices were identified at PORTS, the meteorological programs at PGDP and PORTS were adequate. The radiochemistry laboratories that provided analytical support to the radioactive effluent control program and the radiological environmental monitoring program produced credible, defensible analytical data. The laboratory staff were knowledgeable and experienced, and the laboratories provided the required level of support to the radioactive effluent control and radiological environmental monitoring programs with respect to the required analytical detection levels for the types of samples (process and ambient) submitted to the laboratory. Overall, the data management programs at PGDP and PORTS were determined to be well thought out and effectively implemented.

Report Details

A special inspection team with expertise in Health Physics and Radiological Environmental Monitoring conducted onsite inspections at PGDP and PORTS with the goals to confirm that the certificatee had a clear understanding of the external and internal radiation hazards, and to confirm that the radiological controls and protection measures in place were adequate to protect workers and the public from radioactive material in NRC-regulated areas. In order to perform an assessment of the two overall goals of the inspection, the team's inspection activities were directed by the following objectives:

- Does USEC Understand Work Environment?
- Do USEC's Radiological Controls Protect the Worker?.
- Are External and Internal Dose to USEC Workers Correctly Assessed?
- Is USEC's Environmental and Effluent Monitoring Appropriate?
- Is USEC's Methodology for Measuring Radionuclides Appropriate?

UNDERSTANDING THE WORK ENVIRONMENT

R1 Site Characterization

a. Inspection Scope

The inspectors reviewed site characterization studies, conducted field observations, and discussed, with cognizant personnel, the existing work environment in areas regulated by the NRC.

b. Observations and Findings

b.1 Portsmouth

Issue 13 of the PORTS Compliance Plan required the certificatee to complete radiological characterization and re-posting of leased areas within the site boundary by December 31, 1998. The certificatee conducted site characterization of the leased areas between 1993 and 1997. Areas that remained under DOE oversight were not characterized by the certificatee as part of this effort. Prior to this inspection, the NRC had reviewed the certificatee's completion actions associated with this Compliance Plan issue and had determined the actions to be adequate.

The characterization work included radiation surveys of outdoor and indoor areas and equipment, surveys and smears for fixed and loose contamination, and a search for radioactive materials that were outside of the contamination control areas. Outdoor areas surveyed, estimated at over 560 acres, included roads, parking lots, gravel areas, and grassy areas that had the potential to be contaminated because of the proximity to process buildings or because of past uses. Indoor surveys included building floors, walls up to 8 feet from the floor, structural components, the roofs of process and other buildings with the potential for contaminated effluents, outside wall penetrations in these buildings, equipment surfaces, some equipment seal areas, and other equipment components with a potential for being contaminated, such as motors and air-moving equipment. The inside surfaces of process cells were not surveyed because that equipment was known to be contaminated and access was controlled accordingly, and the same applied to areas above the 8 foot survey level, such as the top parts of cells and other overhead areas.

Also surveyed were an estimated 6,000 employee lockers, desks, eating areas, drinking cups, and an estimated 500,000 tools. Smears with activity exceeding an action level of 1,000 disintegrations per minute (dpm) were subjected to further analysis to determine the identity of the contaminants. Smears from areas with low contamination were sometimes batched to obtain sufficient activity for isotopic analysis. Soil samples were not taken because such sampling did not serve the primary function of the project, which was worker radiation safety, rather than assessment of environmental impact.

The certificatee stated that the major outcome of the characterization effort was to correct the postings at a large number of locations. Many areas found to be contaminated and not posted were posted, and some others, such as lockers and eating areas, were decontaminated. A separate area reduction project, extending over the period 1995 - 1997, was undertaken soon after the start of the characterization project to reduce the total size of contaminated areas onsite. The aim was to improve working conditions by reducing the number of areas that required contamination control measures for entry or during work. Many areas were cleaned of removable contamination and either de-posted or posted as areas with only fixed contamination. However, the inspectors observed that no progress had been made to decontaminate an outdoor area (roughly 500 square yards) immediately outside the southeast corner of X-326 that had become contaminated with up to 100,000 dpm/100 cm^2 of direct beta radiation. This contamination resulted from spillage of lube oil caused by a December 1998 purge cascade cell fire. Data from surveys of this area conducted immediately after the spill could not be obtained at the time of the inspection, and the certificatee took about 28 soil samples from the contaminated areas to conduct isotopic analysis. The certificatee expected to identify ^{99}Tc as the dominant radionuclide. The subject of characterization and cleanup of the southeast corner of X-326 will be tracked as an **Inspection Follow-up Issue (IFI) 70-7002/99013-01** pending review and resolution by the certificatee.

A radioactive materials control program and a tool control program were undertaken during the time period of 1995 -1997. The radioactive materials control program was directed mainly at an inventory of the large number of calibration and check sources onsite and establishing a data base for them. In addition to keeping track of sources, the database provided users with notices to conduct periodic inventory of sources in their possession, done every 6 months, and to leak test their sources, which was required every 6 months, except for plutonium sources, which were leak-tested every 3 months. The tool control program was intended to keep tools from carrying contamination from work areas to clean areas, and also to avoid having to constantly decontaminate tools coming out of contaminated areas. The program involved marking contaminated tools yellow and clean tools green. Workers were trained to keep yellow tools inside of radiologically controlled areas and to refrain from taking green tools into those areas. The certificatee stated that this program had been successful, although there were occasions when tools were found outside of the areas appropriate to the markings.

A review of a random sample of characterization records indicated that the project was extensive; however, the inspectors did not attempt to assess the design of the survey grids nor the technical execution of the surveys. The certificatee stated that contamination was found in many areas of the site, both inside and outside buildings. The cell floors in the process buildings had some contamination, but not extensive, because a contamination control program was in effect before the characterization project. The same was true of the operating floors of the process buildings. Contamination was also found in many outdoor areas onsite, and many such areas were posted contaminated areas, some with fixed contamination and others with both loose and soil contamination.

Many of the chemistry laboratories were also found to be contaminated. The greatest contamination levels were found in Buildings X-700 and X-705, the equipment repair building and the decontamination building. Building X-705 was the more highly contaminated, and the entire building was still posted and operated as a high contamination area.

After an extensive tour of the X-705 Building, the inspectors noted that the postings were adequate. The inspectors did note however, that the certificatee identified two controlled boundaries in X-705 where high airborne concentrations of radionuclides could occur due to cutting and grinding operations. The inspectors noted that the certificatee had erected roughly 10-foot high plastic curtains around one of these airborne radioactivity areas, but there was no roof or top cover over the area within the building. The inspectors questioned the certificatee about the possibility of resuspending and transporting contamination over the curtains into the high contamination area during cutting or grinding operations. The certificatee indicated that a completely enclosed tent could not be erected in this case due to competing criticality concerns, and that while such operations were taking place, portable high efficiency particulate air (HEPA) filter vacuum systems were used to trap resuspended radionuclides and that at least one continuous air monitor (CAM) was placed in a strategical position by health physics (HP) staff outside the plastic curtains. On these CAMs, acute alarms were set at soluble uranium's immediately dangerous to life and health (IDLH) level of 10 mg/m^3, averaged over 6 seconds, and chronic alarms were set at a derived air concentration (DAC) of 1×10^{-10} μCi/ml (98% U-234 and 2% Th-230) averaged over 30 minutes.

Contamination was also present on concrete pads and soil around the X-705 Building. The certificatee stated that much of the contamination most likely came from decontamination of uranium cylinders, which in the past was done with limited contamination control measures. In addition, the outside contamination around the building could have been caused as a result of past decontamination and staging work that was done outside, and partly from contaminated water runoff from inside the building.

Most of the contamination found onsite was uranium, but some thorium and transuranic radionuclides were also found, particularly thorium-230 (^{230}Th), neptunium-237(^{237}Np), and plutonium-239 (^{239}Pu). The source of ^{237}Np, and ^{239}Pu was believed to be the conversion of recycled uranium oxide to UF_6 in the northeast section of X-705 and the feeding of UF_6 cylinders obtained from and contaminated at PGDP. The ^{230}Th radionuclide is a daughter product of ^{234}U, small fractions of which exist in natural uranium. The enrichment process tended to increase the concentrations of ^{234}U in product and in the upper end of the cascade. Although ^{230}Th is not a transuranic (TRU) radionuclide, its radiological hazards and characteristics are similar to those of TRU radionuclides such as ^{237}Np, and ^{239}Pu. Therefore, ^{230}Th was accounted for in the certificatee's radiation protection program as if it were a TRU radionuclide. The ratio of the combined activities of these three isotopes to the total activity, varied widely from location to location, but was generally less than 2 percent except for some isolated locations in Building X-705, where it reached 8 percent for a few samples. Transuranics were also found in the oxide conversion facility, no longer operating, and in some holding ponds, however, neither of these facilities were under NRC oversight.

c.1 Portsmouth Conclusions

Based on the characterization performed and other available data, the certificatee concluded that a 2 percent TRU activity ratio was representative of Buildings X-700 and

X-705, and that 0.5 percent was representative of other areas onsite. The team review of the data indicated that these selections were appropriate and conservative for most situations. The ratios were in use in assessment of internal doses to workers (see Section R6 below). These ratios may not, however, be appropriate for some specific situations where work is being conducted in areas where past surveys have shown ratios higher than the default values. In such cases, job specific ratios, based on analysis of breathing zone air samples, analysis of smears, or other data, may be more appropriate.

The team concluded that USEC's site characterization efforts were effective and resulted in clearly defined and well posted areas that were contaminated or presented other radiological hazards. However, in one instance no progress had been made to decontaminate an outdoor area that had become contaminated from spillage of lube oil during a December 1998 purge cascade cell fire. One **IFI** was identified regarding the cleanup of that contaminated area.

b.2 Paducah

Site characterization was very similar to the effort at PORTS, as discussed above, much of the work was done by the same contractors. Issue 10 of the PGDP Compliance Plan required the certificatee to complete all radiological characterization and posting of leased areas within the PGDP boundary by November 30, 1997. Prior to this inspection, the NRC had reviewed the certificatee's completion actions associated with this issue and had determined the actions to be adequate.

The certificatee began the site characterization in February 1995 and completed it in November 1997. Decontamination was not part of the project, but some decontamination was undertaken when appropriate. Buildings were surveyed using direct contamination surveys, smears, and exposure rate measurements. Surveys included floors, structural supports, and equipment and walls up to a height of 6 feet. Overhead areas were not surveyed except where necessary because of frequent worker access. These un-surveyed overheads were controlled as contaminated areas. Similarly, cell enclosures were not surveyed. These were also controlled as contamination areas. Lockers, desks, tools, eating areas, and many outdoor areas were surveyed, and a tool control program was established. An effective source control program had already been in operation before the characterization project. The surveys resulted in posting many areas as contaminated areas that had not been previously controlled, and many areas were found that did not need to be posted. The certificatee's characterization for areas outside buildings involved direct surface contamination measurements. Prior to the characterization, there were about 13 buildings onsite that were radiologically posted for contamination. Following the characterization, 29 buildings were posted for radiological contamination. Surveys were conducted on 250 acres of floor space inside buildings and roughly 460 acres of outside areas including 27 miles of roadway. After the characterization effort was completed, the certificatee returned to DOE (i.e., deleased) around 215 acres of outside areas that were not needed for the purpose of enriching uranium. A significant fraction of these areas included grassy areas contaminated by legacy operations.

A contaminated area reduction program was initiated concurrently with the characterization effort, with the object of reducing the number and size of posted contamination areas within buildings to increase work efficiency. The certificatee stated that about one million square feet of contaminated areas inside buildings were cleaned and de-posted. The inspectors noted that as a result of the site characterization, the certificatee had identified as contaminated and roped off several small areas outside buildings; some as small as a few square feet. The inspectors were informed that there were no immediate plans for decontaminating these “small pockets” of contamination and that these areas would likely be decontaminated at the time of decommissioning. The inspectors considered this approach might be contrary to the ALARA principle and recommended that a follow-up review of this item be conducted by Region III. The subject of decontaminating small pockets of contaminated areas will be tracked as an **Inspection Follow-up Issue (IFI) 70-7001/99013-01.** The certificatee also stated that isotopic analysis of samples was not within the scope of the project, although some isotopic analyses were conducted when samples with unexpectedly high activity levels were obtained. The inspectors noted that the certificatee had identified as contaminated some leased areas directly adjacent to DOE’s clean scrap yard (C746-C1). However, there was no isotopic data available for these areas and the certificatee did not have a program to monitor for any migration of contamination from DOE storage areas to USEC-leased areas.

The inspectors were provided results of isotopically analyzed concrete floor samples obtained as a result of the seismic upgrades in cascade Building C-331 and C-335. The inspectors noted that other than some small positive measurements of uranium (a few picocuries per gram (pCi/g)), all TRU radionuclides were determined to be less than the minimum detectable activity (MDA) of 1 pCi/g.

Smear and air samples obtained in 1991 and 1992, along with unknown quality control (QC) samples were analyzed by Oak Ridge National Laboratory (ORNL). According to the certificatee, the results reported by ORNL for the QC samples showed wide variance from the known activity. Therefore, these sample results were declared invalid and not considered in determining location-specific TRU concentrations. Air and wipe samples were also taken in 1993 and analyzed by the PGDP laboratories, but no QC samples were submitted. Since the uranium enrichments determined by the PGDP laboratories were not representative of the locations from where the samples were taken, the certificatee also declared these samples invalid and did not consider the results in determining location-specific TRU concentrations.

Additional site TRU and other radionuclide characterization programs were conducted, one in 1992 and a second in 1993, by Oak Ridge Institute for Science and Education (ORISE). The projects consisted of radiation and contamination surveys of buildings thought to most likely contain transuranic contamination, and included the decontamination building (C-400) as well as the feed and withdrawal buildings. Samples with high levels of alpha (α) or beta (β) activities were analyzed for ^{99}Tc and transuranics. The surveys found many contaminated areas that were not posted as such, containing both fixed and removable contamination. Isotopic analyses also showed the presence of ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, $^{239/240}$Pu, and ^{99}Tc in many locations in all of the buildings surveyed. Most of the samples showed fractional TRU activities on the order of 1 percent or less, with a mean level of 3.1 percent and a maximum of 9 percent in the decontamination building, and a mean of 0.7 percent and maximum of 3.9 percent for other buildings.

The certificatee selected a value of 8 percent for the TRU fractional activity as representative of the site as a whole. The reason given was that this level was representative of the decontamination building, and because most pieces of equipment from all parts of the plant were brought into this building for decontamination, then this value was also representative of the other areas of the plant. In addition, it was also representative of activities found in other areas of the plant.

An earlier attempt at characterization was made in 1991 by using the TRU body burdens of workers as an indicator of the types and levels of contamination at the site. A group of 16 workers was chosen based on the fact that they had been working for some time at the site and were engaged in work that had the highest potential for intakes. The workers were sent offsite for whole body counting and also provided fecal samples for isotopic analyses. All tests showed no TRU activity.

The certificatee stated that there was a large body of data which suggested that the levels of TRU at the site had decreased to negligible levels since the characterizations described above were conducted. Air sampling data collected since 1997 (a total of well over 50,000 air samples) showed only 27 samples that indicated gross alpha air concentrations that exceeded 10 percent of the site derived air concentration (DAC). The DACs for PGDP are based on the radioactive nuclides present at the site and are listed in SAR Table 5.3-11. The site DACs are based on the more limiting of 5 rem Committed Effective Dose Equivalent or 50 rem tissue or organ limit. These 27 samples were analyzed for TRU, and nearly half showed ^{237}Np, ^{239}Pu, and/or ^{241}Am. In general, the concentrations were small and the average transuranic activity ratio for all the samples was about 2 percent.

In addition to the above, UF_6 samples were routinely analyzed for transuranics, and showed only traces of TRU activity. Analyses of product and tails samples also showed the same trace amounts, and wash liquids from cylinder cleaning also showed trace amounts, except for two samples which showed 1.6 percent and 3.9 percent ^{237}Np. Samples from the decontamination building filtrate tanks showed higher levels of transuranics, all ^{237}Np, with the levels generally being on the order of 2 to 3 percent.

In 1994, per the ORISE characterizations of 1992 and 1993, the certificatee concluded that an 8 percent TRU activity ratio was a representative maximum value for the site, and that number was included in the SAR (Table 5.3-11) for the site. However, based on more recent data, and for the purposes of assessing internal dose via bioassay urine analyses, the certificatee stated that the 8 percent TRU was not representative of current site conditions and, for routine bioassays, a value of zero percent TRU was being used. Even though the 8 percent figure was not used to calculate internal doses as described above, it was used to calculate a site-specific DAC, using an activity ratio of 92 percent uranium and 8 percent thorium (^{230}Th). The certificatee stated that the activity ratio for ^{230}Th (a decay product of ^{234}U) was used in calculating the site-specific DAC because it was found in many places onsite and was a conservative surrogate for TRU. The ^{230}Th DACs (Class W and Y) were reasonably similar to the DACs of the Pu isotopes and ^{237}Np isotope. Also, ^{230}Th was found in most samples that contained other TRU, and in generally much higher concentrations; therefore, the use of ^{230}Th for calculating the site DAC appeared reasonable.

c.2 <u>Paducah Conclusions</u>

The data reviewed during this inspection showed that there was a lot of current data on the isotopic composition of process stream materials, such as the UF_6, the feed, and the product streams. There was also much data on air samples drawn over the past few years, most of which showed very low levels of airborne activity. Although the flow stream data showed that the levels of TRU in these streams were very low, existing only in trace amounts, the air sampling data could not be used to reach a similar conclusion regarding airborne activity. A total of well over 50,000 air samples were collected and analyzed for gross activity over the past few years, and most had gross activities below the level that required isotopic analysis. Only 27 of the total air samples exceeded the level requiring analysis for isotopic content. Many showed the presence of TRU, although most showed only very low TRU levels. Based mainly on these air samples, the licensee concluded that the appropriate level of TRU to use in the dose assessment of routine urinalysis data was zero percent. However, the team did not believe that this conclusion was warranted for several reasons. A significant fraction of the air samples that were analyzed for isotopic content did show the presence of TRU. If that fraction of analyzed samples that did show TRU was applied to all the air samples taken, most of which were not analyzed for isotopic content, then it must be concluded that a large number of these samples must have contained TRU. In addition, a large fraction of the air samples were general area air samples. Such samples measure the airborne activity in the general area in the building at the location of the sampler. Such a location may be quite far from the source of the airborne activity, and the activity measured will therefore be diluted from its source level by possibly several orders of magnitude. Any activity that may have been present at the source may have fallen to below detection levels after dilution.

The team's review of the available data also showed that relatively few smear samples had been analyzed for isotopic content, the number that was analyzed being too small to provide a reliable assessment of the distribution and magnitude of the levels of TRU in different areas of the plant. The conclusion reached by the team was that the historical data suggested the wide-spread but low-level presence of TRU, and that there was insufficient current, reliable, and representative data to change the conclusions drawn from that historical data. The use of zero percent TRU in dose assessment, although it may be warranted, was not sufficiently supported by the available data, and was also contrary to the value given in the SAR. The decision to exclude TRU from some internal dose assessments is further discussed below in Section R6.b2; and, an IFI (70-7001/99013-06) is identified in that section to track the certificatee's review and resolution of this issue.

The team concluded that USEC's site characterization efforts were effective and resulted in clearly defined and well posted areas that were contaminated or presented other radiological hazards. However, there were a number of "small pockets" of contaminated outside areas that the certificatee had no plans for decontaminating. The team identified a concern regarding that lack of planning. One **IFI** was identified regarding plans for cleaning up small pockets of contaminated areas.

RADIOLOGICAL CONTROLS TO PROTECT WORKERS

R2 Health Physics Organization, Qualifications, Training, and Boundary Controls

a. Inspection Scope

The inspectors assessed the qualifications, training and effectiveness of Health Physics personnel, and site boundary controls through review of records, interview of personnel, and observations of field activities.

b. Observations and findings

b.1 Portsmouth

The radiological controls program onsite was the responsibility of the Health Physics Group, under the general direction of the Radiation Protection Manager (RPM). The

Health Physics Group was one of six groups within the Production Support organization. The Managers of these six organizations reported to the Production Support Manager, who in turn reported to the Enrichment Plant Manager. The Enrichment Plant Manager reported to the site General Manager, the highest management position onsite. The reporting chain for the radiological controls organization was therefore the RPM, Production Support Manager, Enrichment Plant Manager, and Site General Manager. The inspectors noted that this organization appeared to present potential conflict of interest problems because the RPM reported to a production manager. The certificatee stated that there had not been any conflict of interest problems to date, and that the health physics group had stop work authority if any unsafe radiological condition was identified, an authority that had been exercised on several occasions in the recent past. The RPM also stated that he had the option of bypassing his two line supervisors and going directly to the site General Manager in case of difficulties, but there had been no occasion to exercise this option. During the onsite inspection, the inspectors did not observe or identify any instances that might indicate a conflict of interest arising from this reporting chain.

The Health Physics Group, consisting of a staff of 56 (USEC employees), was organized into three subgroups: Technical Support, Health Physics Operations, and Dosimetry. The Technical Support group consisted of four persons who provided technical oversight for the program, established procedures, reviewed data, maintained the radioactive materials database, calculated internal and external doses resulting from routine and unusual exposure situations, oversaw the external and internal dosimetry programs, and generally provided technical health physics advice and support to the rest of the organization and to other groups onsite.

The Health Physics Operations group, with a total of 44 technicians and supervisors, provided the radiological controls support for onsite radiological work. This group was divided into 3 sub-groups: Laboratory, Maintenance, and Decontamination; Shift Staffing and Training Support; and Cascade Operations. Technicians were assigned specific plant areas for which they were responsible to ensure proper radiological controls. The technicians attended daily planning meetings for their areas, discussed job details for upcoming jobs and wrote Radiation Work Permits (RWP), and ensured that radiological controls were properly carried out. The technicians were periodically rotated to other areas of the plant, and they also did shift rotation. Discussions with health physics technicians during the inspectors' tours of the plant demonstrated that the technicians were knowledgeable of the activities that were conducted in their assigned buildings, were aware of potential hazards, and knew the radiological conditions in these areas.

There were three technician classifications: junior, senior, and technologist. Junior technician was an entry level position requiring a high school diploma and no experience. A senior technician required a high school diploma, 2 years of radiological experience, and 3 months onsite. These qualifications met the standards for a senior radiation protection technician specified in ANSI/ANS 3.1, 1987. A technologist required a minimum of 7 years radiological experience, and 6 months onsite. All health physics supervisors were technologists. At the time of the inspection, there were no junior technicians onsite.

Technicians must attend a minimum of 10 continuing training sessions per year, each session lasting 2 - 4 hours. Topics discussed during these sessions included recent events, procedure changes, new equipment and methods, and other current issues. The training was provided by health physics supervisors and occasionally by a member of health physics management. All technicians also received an initial 24 hour hazardous material training course, and an annual 8 hour refresher.

Technicians were required to re-qualify every 2 years by taking a comprehensive written examination. An exception to this rule applied to technicians qualified by the National Registry of Radiation Protection Technicians (NRRPT), who did not need to re-qualify. At the time of the inspection, all technologist working onsite were NRRPT qualified, as well as some senior technicians. Promotion to a higher level required both a written examination and an oral board.

A review of randomly selected resumes of 36 health physics technicians showed that most had many years of experience at the site, but many had little or no experience at other facilities. Most of those with experience outside of PORTS worked previously in the Navy, but some worked in the nuclear power industry. All the technicians reviewed exceeded the minimum education and experience requirements for their current positions, most by a wide margin. The qualifications of the Radiation Protection Manager where also reviewed and found to exceed the minimums for this position although, as was the case with a large number of the technicians, all of the experience was gained at PORTS.

The inspectors reviewed the radiological postings and boundary controls at PORTS. Generally, the certificatee maintained adequate radiological postings and effective radiological boundary controls. The certificatee controlled access to areas where radiation or radioactive material/contamination may be encountered via the use of RWPs, and setting up physical boundaries and postings. The certificatee used 1000 dpm/100 cm^2 removable and 5000 dpm/100 cm^2 fixed as criteria for determining contaminated areas. According to the PORTS SAR, these criteria applied to uranium and technetium with transuranics amounting to less than or equal to 2 percent of the activity. Even though the SAR contained criteria for transuranics above 2 percent, according to the certificatee, these were not used for radiological protection purposes since there was no indication that transuranics existed above the 2 percent level in leased areas at PORTS.

The certificatee used Berthold hand and foot monitors in conjunction with hand-held survey monitors (friskers) for the monitoring personnel leaving contaminated areas. The inspectors noted that there was a lack of care on the part of some workers to adequately monitor themselves using the friskers. Considering the certificatee’s procedural requirement of scanning at 1-2 inches per second, the inspectors estimated that it should take 3-4 minutes to complete monitoring of an averaged sized worker. However, the inspectors observed some workers exiting the boundary control stations frisked themselves in less than one minute (subsequent to the inspector’s observations, a USEC HP supervisor directed the workers to resurvey using the proper technique). The certificatee was in the process of installing Eberline PCM2 whole body monitors at the boundary control stations. These type of monitors will reduce the time it takes to conduct personnel monitoring and also significantly reduce the use of hand-held instruments for personnel monitoring.

The inspectors observed boundaries established in the process buildings, UF_6 feed and withdrawal areas, laboratories, decontamination building and outside areas. The inspectors observed some unsecured items such as hoses, flexible instrument air lines, and solid items staged across boundaries (all noted discrepancies were corrected or appropriate actions were begun to correct the deficiency at the time of the observations). This could lead to the spread of contamination across these boundaries. The inspectors also noted that improvements were warranted in the area of general housekeeping in process buildings. The inspectors noted that an area in the operating floor of Building X-326 was not clearly marked as a DOE Material Storage Area (DMSA). While in Building X-326, the inspectors were required to review and sign a general entry RWP in the building's Area Control Room. That RWP referred to another RWP for access to cell floor areas located south of Column 82. The second RWP required an additional pair of latex gloves in the contaminated areas south of a Column 82; however, the inspectors observed several workers crossing the unmarked floor location believing only the first RWP applied to their tasks. The physical location for the additional RWP requirements was not marked in the field and obviously resulted in the worker confusion. The certificatee acknowledged the confusing RWP requirements and made appropriate corrections.

c.1 Portsmouth Conclusions

The team concluded that the qualifications, training and effectiveness of Health Physics personnel were adequate. Although generally effective, several weaknesses were identified in the boundary control program for contaminated areas. Inclusion of the radiation protection program within the production organization has the potential to cause subtle and hard to detect biases in the program's effectiveness and its ability to implement a high quality safety program; however, no such effects were identified over the course of this inspection.

b.2 Paducah

The Health Physics Department was under the direction of a Health Physics Manager, who reported to the Production Support Manager. A Radiation Protection Manager (RPM) also reported to the Production Support Manager. The RPM was responsible for the administrative aspects of the Health Physics organization and had no line organization reporting to him, while the Health Physics Manager was responsible for the Operational Aspects.

The Production Support Manager reported to the Enrichment Plant Manager, who in turn reported to the General Manager onsite. As was the case at PORTS, the Health Physics organization was under a production manager, thus creating a potential conflict of interest. However, no evidence of such potential conflict was seen during the inspection.

The Health Physics group was divided into a Cascade Group, a Technical/Cylinder Group, and a Balance of Plant Group. There were about 70 technicians in the organization, many of whom were contractors. A review of a random sample of 48 technician resumes showed that the median experience at PGDP was 7 years, with a mean also of 7 years, and the median level of experience prior to working onsite was zero years, with a mean of 5 years. Many of the technicians with outside experience had very long health physics experience either in the Navy or in the nuclear power industry.

Technician classifications and required experience for each level were comparable to those at PORTS, as described above, with some minor differences. Newly hired experienced technicians were provided with a self study manual and then must pass an examination, followed by on-the-job training. Junior technicians were offered an extensive training course that included basic theory and applied health physics topics. All technicians were required to re-qualify every 2 years.

Technicians were assigned to functional plant areas or buildings, and they were responsible for implementing radiological protection in their areas. Discussions with technicians during plant tours showed that the technicians were well informed and aware of the hazards in their areas, and of the ongoing jobs and associated protection measures and requirements.

The inspectors reviewed the radiological postings and boundary controls at PGDP. The boundary control program was similar to the one established at PORTS, and similar weaknesses were noted by the inspectors. Similar to PORTS, the certificatee used 1000 dpm/100 cm^2 removable and 5000 dpm/100 cm^2 fixed as contamination criteria for all leased areas at PGDP. According to the PGDP SAR, these criteria applied to uranium and technetium with transuranics amounting to less than or equal to 2 percent of the total activity. The certificatee did not provide the inspectors with a valid justification for using these criteria even though stricter criteria were contained in the PGDP SAR for transuranics above 2 percent. The subject of using SAR criteria for radiological posting and boundary control will be tracked as an **Inspection Follow-up Issue (IFI) 70-7001/99013-02** pending review and resolution by the certificatee.

The certificatee used Berthold hand and foot monitors in conjunction with hand-held survey monitors (friskers) for monitoring personnel leaving contaminated areas. The inspectors noted that there might be a lack of care on the part of some workers to adequately monitor themselves using the friskers. The certificatee was in the process of installing Eberline PCM2 whole body monitors at some boundary control stations. These will reduce the time it takes to conduct personnel exit monitoring and also significantly reduce the use of hand-held instruments for personnel monitoring.

The inspectors observed boundaries established in the process buildings, UF_6 feed and withdrawal areas, laboratories, decontamination building and outside areas. The inspectors noted some unsecured objects such as hoses, metal beams, equipment cover plates, chipped concrete debris, etc. staged across boundaries. Although not observed, these types of practices could lead to the spread of contamination across posted boundaries. The inspectors also noted one of the four boundary ropes to be down between two autoclaves in C-333A which was considered to be a contaminated area, and improperly placed "Radiation Area" postings in Building C-360. These postings were corrected by the certificatee in the presence of the inspectors. The inspectors also noted that improvements were warranted in the area of general housekeeping.

Not many soil excavations occur in leased areas at PGDP. However, the inspectors noted two excavations outside the two PGDP UF_6 feeds, C-333A and C-337A. The inspectors were informed that a general directive was issued, that required informing the plant's HP group of any impending excavations or disturbances of outside areas, and the inspectors noted that for the two excavations observed, the HP group had been properly notified. The workers had also been directed to inform HP if unexpected items were unearthed during excavations.

The certificatee had established an extensive respiratory protection program at PGDP. The use of respirators, which was controlled by PGDP's Industrial Hygiene group, served a dual purpose of providing chemical safety as well as radiological safety for workers. Over 1000 workers were qualified to don respirators at PGDP. The certificatee informed the inspectors that respiratory protection was required if significant resuspension could occur in an area or if cutting or grinding operations were to be done in an area that had contamination levels exceeding 50,000 dpm/100cm^2 beta/gamma. However, the inspectors noted that the certificatee had no specific procedural requirements concerning the use of respirators. The inspectors were informed that a June 1997 audit report had identified a similar finding. An assessment and tracking report was initiated at the time of NRC's inspection to generate formal procedural requirements with a completion date of October 27, 1999. One **IFI** was identified regarding criteria used for radiological postings.

c.2 Paducah Conclusions

The team concluded that the qualifications, training and effectiveness of Health Physics personnel were adequate. Although generally effective, several weaknesses were identified in the boundary control program for contaminated areas. Inclusion of the radiation protection program within the production organization has the potential to cause subtle and hard to detect biases in the program's effectiveness and its ability to implement a high quality safety program; however, no such effects were identified over the course of this inspection.

R3 Radiation Worker Training

a. Inspection Scope

The inspectors reviewed available training material and spoke with cognizant personnel in the health physics organization regarding the scope and depth of training provided to the work force.

b. Observations and Findings

b.1 Portsmouth

Two levels of radiation worker training were provided: Rad I, consisting of 12 hour classroom lectures and practical factors, and Rad II, consisting of 16 hours of classroom training and practical factors. The classroom material was substantially the same for both courses, and differed mainly in the extent of coverage of contamination control topics and in the practical factors. General Employee Training (GET) was required before attending either Rad I or Rad II training.

Rad I was intended for people who may enter radiation areas or contamination control zones (CCZ), but not contaminated areas. A CCZ was generally considered to be a clean area that surrounded one or more contaminated areas and acted as a buffer zone for those areas. Rad I training was generally given to clerical staff, engineering staff, and others who might pass through radiation areas or CCZs or occasionally observe work in such areas. Practical factors training was limited to the use of shoe covers and gloves. Issuance of a thermoluminescent dosimeter (TLD) required completion of at least Rad I training.

Rad II training was intended for personnel who may enter and work in contaminated areas. The lecture part of the training expanded the discussion on contamination controls, and practical factors training included the use of full protective clothing, surveys, and exit frisking.

Both Rad I and Rad II workers required refresher training every 2 years. This involved taking a written examination followed by an 8 hour training session to provide updates on developments affecting radiation protection onsite. At the time of this inspection, there were about 330 Rad I and 1530 Rad II employees onsite.

A review of the student handout booklets given to Rad I and Rad II trainees showed that some parts of the material were confusing and may be inadequate to convey the information to the trainees. In addition, the information in some parts was erroneous, such as an incorrect statement of a regulatory requirement, or was stated in a manner that provided the student with an incorrect impression of the actual requirement. The discussion of biological effects minimized the risk of radiation to the point where it became difficult to justify any radiological protection efforts or ALARA, and in some cases, statements were made about radiation risks that could not be supported given current knowledge in that field. There was also little site specific information regarding TRU in the manuals. The applied sections of the text, such as the sections on contamination control, ALARA, and radiation work practices were of significantly better quality than the sections discussing theory and regulations. At the end of the team's onsite inspection, the certificatee had initiated a review of the current training material, and the subject of accurate and complete training material will be tracked as an **Inspection Follow-up Issue (IFI) 70-7002/99013-02** pending review and resolution by te certificatee.

c.1 Portsmouth Conclusions

The assessment of radiation worker training showed that some of the material provided to the trainees was outdated and therefore incorrect. The training material on radiological hazards minimized these hazards, thus undermining the intent of the training, which was to enable workers to recognize and avoid radiological hazards in the field and effectively participate in their own "ALARA." That goal was also undermined by the near complete absence of descriptions of site specific radiological hazards, including their nature, magnitude, history, and methods of avoidance and control, making it difficult for the worker to participate in implementing good radiological work practices where and when appropriate, and also weakening the worker's ability to recognize hazardous radiological situations. There also appeared to be little re-enforcement of the formal training between required training sessions. One **IFI** was identified regarding the certificatee's review of current radiation worker training material.

b.2 Paducah

Radiation worker training requirements were the same as those at PORTS, but the training material was different. A review of the student handout materials for both Rad worker Levels I and II showed that the material was well organized and well presented. However, the radiological risks were minimized in the same manner as at PORTS. There were also some inaccuracies, such as an incorrect definition of the whole body that did not correspond to that in 10 CFR Part 20.

Some concepts that were important to radiological protection in the plant were not presented clearly in the text. As an example, the site defined areas called Contamination Control Zones (CCZ) which were supposed to act as buffer zones between clean and contaminated areas. In practice, entry into and exit from these areas was conducted as though the area was slightly contaminated: some precautions against contamination were required, some protective clothing such as gloves may be required, and frisking out of the area was required. However, the description of such areas in the training material was quite difficult to understand, and did not prepare the worker for such an environment. It was noted that this observation was drawn entirely from a review of the training material, as the inspectors did not attend any onsite training sessions.

A review of the training material also showed that there was very little discussion of site specific radiological hazards, such as the possible sources of radiation exposure, contamination, and airborne activities, the nature of such radiation fields and contaminants, and the ways in which they may arise. Random discussions with workers at the site during this inspection showed that many workers had very little understanding of the radiological hazards or the sources, and some did not take appropriate care in implementing required radiological controls, particularly when frisking out of CCZs. On the other hand, jobs that were under direct health physics supervision were found to be well controlled, and all appropriate radiological precautions were implemented. It was not possible to make a direct correlation between the lack of site specific material in training and the observed lack of worker knowledge noted above. At the end of the team's onsite inspection, the certificatee had initiated a review of the current training material, and the subject of accurate and complete training material will be tracked as an **Inspection Follow-up Issue (IFI) 70-7001/99013-03** pending review and resolution by the certificatee.

c.2 Paducah Conclusions

The team's conclusion was that the radiological training material was generally good. As at PORTS, field observations by the inspectors revealed that some workers had little knowledge and understanding of radiological hazards and of appropriate radiation protection procedures under specific field conditions. Some were also observed to neglect implementing required contamination control procedures. As discussed above for the PORTS site, the training material on radiological hazards minimized these hazards, thus undermining the intent of the training, which was to enable workers to recognize and avoid radiological hazards in the field and effectively participate in their own "ALARA." That goal was also undermined by the near complete absence of descriptions of site specific radiological hazards, including their nature, magnitude, history, and methods of avoidance and control, making it difficult for the worker to participate in implementing radiation good practices where and when appropriate, and also weakening the worker's ability to recognize hazardous radiological situations. There also appeared to be little re-enforcement of the formal training between required training sessions. One **IFI** was identified regarding the certificatee's review of current radiation worker training material.

ASSESSMENT OF EXTERNAL AND INTERNAL DOSE TO WORKERS

R4 Air Sampling Program

a. Inspection Scope

Integral to the assessment of dose to workers is an effective air sampling program, which provides needed data for ensuring worker protection while in an airborne environment. The inspectors reviewed data from the certificatee's air sampling programs, observed field use of instrumentation, and discussed program objectives and results with cognizant personnel.

b. Observations and Findings

b.1 Portsmouth

The air sampling program was used mainly to monitor for trends in airborne activity and to estimate current transuranic activities, to classify or declassify airborne radioactivity areas, and to provide supporting information in cases where there were questions regarding the validity of bioassay results. The air sampling results were not normally used to assess intakes.

Air sampling at the site could be classified into routine and job related or special sampling. The routine program used a large number of stationary, low volume, continuous air samplers that consisted of a sampling head with a membrane filter that was attached to a building vacuum manifold, with the flow rate maintained at 20 liters-per-minute (L/min). The filters were changed daily, except on weekends. The air flow through the filter could be regulated using a local flow valve. No flow gauge was provided at the sampler, but flow was measured using a portable gauge whenever the filter was changed. The filter head was removed from the vacuum pipe and the gauge was placed in series between the sampler and the pipe. This provided flow rate readings at the beginning and end of each sampling period, although the reading may not be very accurate because the flow resistance changed slightly when the gauge was introduced into the flow path, and the flow in the intervening period was also dependent on the building's vacuum system.

The filters were counted on a gas flow proportional counter for 20 minutes, given what the certificatee estimated to be a detection sensitivity of $2x10^{-14}$ microcuries-per-milliliter (μCi/ml) alpha (α) and $4x10^{-14}$ μCi/ml beta (β) activity. The α sensitivity corresponded to about 0.02 percent of the site DAC of $1x10^{-10}$ μCi/ml. The certificatee stated that this fixed sampling system was being replaced gradually by portable, self contained, low volume samplers, and a few of those were already in use. The air data from this system was not representative of the concentrations to which workers in the area may be exposed, and the data was used only to trend airborne activity in the general area, and occasionally to help narrow down possible locations of small leaks of radioactive material.

Job related or special sampling was conducted using three main types of sampling equipment: High volume air samplers, continuous air monitors (CAM), and breathing zone samplers (BZA). High volume samplers were used to collect a large volume of air in the work area in a short period of time, and those samplers were used mainly to change the classification of a work area, possibly permitting the removal of respiratory protection equipment. The flow rates used with these samplers were 850 - 1100 L/min, and the sample collection time was typically about 10 minutes. There were about 15 high volume samplers in use at the site. The certificatee estimated that field counting of the filters was capable of achieving a sensitivity of about $1x10^{-11}$ μCi/ml α and $4x10^{-11}$ μCi/ml β. The α sensitivity represented 10 percent of the site DAC.

High volume samplers at the site were fitted with a device called an annular kinetic impactor, which was a one stage inertial collector designed to collect airborne particulates above a certain specified size set by the operating parameters of the device. The theory was that freshly formed radon daughters were very small (0.001 - 0.01 micrometers (μm)), whereas airborne radioactive particles were much larger. The impactor deposited the larger particles on a collecting plate coated with a sticky substance, but allowed the smaller particles to pass through the instrument. This permitted samples to be quickly evaluated in the field without having to wait for decay of the radon daughters, which may interfere with counting and give a false high indication of airborne activity. According to the manufacturer's specifications, the device will collect all particles above 0.4 μm in aerodynamic diameter.

The inspectors noted that, since the size distribution of airborne radioactive particles had not been measured at the site, it was possible that the impactor was passing, uncollected, a significant fraction of the airborne radioactivity of interest. The certificatee stated that a test was conducted in 1994 in which a set of air samples was collected from the same area and at the same time using both an impactor and an air sampler without an impactor. The results showed that the activities collected without an impactor were a factor of 2 higher than those collected with the impactor. Based on this test, the procedure currently was to multiply all air sample results obtained using an impactor by a factor of 2.

The second instrument used in job specific or special monitoring was the CAM, and there were 25 CAMs in use at various locations onsite. The CAM was equipped with a special inlet screen that removed freshly formed radon decay daughters, and the airborne particles were collected on a membrane filter, which was changed at least three times per week. The flow rate through the CAM was usually about 30 - 40 L/min. The filter was monitored by a Passivated Implanted Planar Silicon (PIPS) detector which collected an alpha spectrum of the deposit on the filter. A microprocessor striped away the three major alpha peaks emitted by radon daughters that were attached to dust particles and were not removed by the screen (8.78, 7.68, and 6.05 MeV), leaving the contributions from other airborne activity, such as uranium (4.6- 4.8 MeV) or plutonium (5.15 - 5.5 MeV). The CAM was used mainly to give a real time alarm in case of a rising air concentration during ongoing radiological or other work. The alarm set point was usually 10 percent of the site DAC. The microprocessor also provided other useful data, such as the air concentration of alpha activity, and trending plots showing the variation of activity in the air over periods of several days. CAMS were required to be in use in the autoclave building, in Building X-705, and in some locations in the process buildings.

Breathing zone samplers (BZAs) were used to provide information on the air concentration at the worker's breathing zone, and the data might be useful in assessing unexpected bioassay results. The samplers used membrane filters at a flow rate of 4 - 6 L/min. BZAs were required to be used when working in the calciner area, where the airborne uranium was in a form that differed from that in the rest of the plant. The uranium in the calciner was in oxide form, classified as International Committee on Radiation Protection (ICRP) Class Y, and presented a different type of hazard to the workers than the soluble uranium found in other parts of the plant.

Air sampling procedures required that all samples that show concentrations exceeding 10 percent of the site DAC be sent to the analytical laboratory for an isotopic analysis and evaluation by the health physics technical staff. The certificatee's statistical data showed that about 1,000 air samples were collected per month. About 6 percent of these, or 60 samples per month, were evaluated because the samples exceeded the 10 percent DAC action level.

The inspectors' review of the air sampling procedures noted that the procedures sometimes did not give sufficient detail to enable unambiguous implementation. For example, the procedures did not indicate that the impactor on the high volume sampler had a 50 percent collection efficiency and that adjustments must be made to the counting data. This information, however, was included in a reference card, carried by all health physics technicians, that gave equations and factors to be used in calculating air concentrations in the field. The procedures were also somewhat out-of-date in that the procedures did not always reflect exactly the way the air sampling program was being implemented. The technical basis document for the air sampling program was also somewhat out-of-date. Information on methods of implementing various aspects of the program was sometimes scattered over several procedures, making implementation cumbersome. The certificatee was aware of all these inspector observations, and was in the process of revising the technical basis document and the procedures. The subject of revising technical basis documents and procedures will be tracked as an **Inspection Follow-up Issue (IFI) 70-7002/99013-03** pending review and resolution by the certificatee.

c.1 Portsmouth Conclusions

The teams' overall assessment was that the air sampling program was adequate for the type of airborne radiological hazards encountered at the site, but the documentation describing that program, including operating procedures and the technical basis document, were out of date and did not always reflect the manner in which the program was actually implemented, nor did such documents clearly describe the functions of each of its components in radiation protection, isotopic ratio determinations, and dose assessments. The air sampling program provided a sufficient number of samples to enable monitoring and control of general area airborne activities, and the program also provided adequate detection sensitivity for the isotope mix found at the site. One **IFI** was identified.

b.2 Paducah

The air sampling procedures at PGDP closely paralleled that at PORTS, as described above, with some important differences. The low volume sampling system did not use fixed nozzles and building vacuum, but instead used stand alone units, equipped with vacuum pumps and gauges. There were about 160 of these units distributed throughout the site, and filters were changed daily. The filters in use with this system were cellulose fiber filters, which had good collection efficiency, but the certificatee used a 0.56 correction factor to account for self absorption of the alpha particles that were embedded in the matrix of the filter. The inspectors noted that such high correction factors for self absorption were generally considered excessive, and that other types of filter paper might be appropriate.

The certificatee also used a single stage impactor collector head with the high volume air samplers that was identical to that used at PORTS. However, the certificatee assumed a collection efficiency for the airborne activity of 95 percent, rather than the 50 percent used at PORTS. The certificatee stated their belief that the efficiency factor was appropriate, and that it was based on data provided by the manufacturer as well as on data found in some NRC guidance documents. The inspectors noted that the ICRP used a default particle median size of 5 μm for industrial aerosols, and for this size aerosol the impactor would have an efficiency of effectively 100 percent. However, it was not clear whether the aerosols found at the PGDP site fit that default assumption, and this was especially the case in view of the results obtained at PORTS. The subject of collection efficiency will be tracked as an **Inspection Follow-up Issue (IFI) 70-7001/99013-04** pending review and resolution by the certificatee. In addition to the question of collection efficiency, the certificatee used a collection planchette that was coated with a layer of grease to collect the particles in the impactor. Although this method of collection was efficient, it caused the collected particles to become embedded in the grease, which in turn required an alpha self-absorption correction factor of about 50 percent when counting the planchettes.

The certificatee also did not have a means for real time monitoring of airborne radioactivity, such as a CAM, and there were no means to provide timely warning of rapidly changing airborne radioactivity conditions. The certificatee stated that CAMs had been used in the past but too many false alarms were received due to the presence of radon in the air. The inspectors noted that while this may be a problem, a careful evaluation of work involving the potential for unexpected and rapid rises in airborne activity was needed. The inspectors noted that this was particularly important because some personnel onsite tended to regard the high volume samplers as close to real time methods of assessing airborne activity, but in fact the samplers are not capable of providing such rapid response. Site procedures required a minimum volume flow per sample of 400 cubic-feet (ft^3) for high volume samplers and, at a flow rate of 50 ft^3/min, it would require about 8 minutes to collect an air sample. Adding about 4-5 minutes to remove the filter and perform a field assessment of the activity results in a minimum assessment time of about 12 minutes.

c.2 Paducah Conclusions

The team concluded that overall, the air sampling program was adequate for the intended purpose, which was mainly trending of airborne activity and establishing controls such as posting and respirator removal. However, the role of the air sampling program in the overall protection and dose assessment functions onsite was not adequately documented in the technical basis document. This was particularly true in view of the fact that the results of that program were being used as the basis for certain important internal dose assessment assumptions. The correlation between general air sampling results and the results of smear and breathing zone air sample isotopic analyses were also not clearly established. Such correlations are necessary if the air sampling results are to be used as indicators of the relative isotopic abundances of uranium and TRU onsite. One **IFI** was identified regarding the high volume air sampler collection efficiencies used at PGDP versus the efficiency used at PORTS for the same equipment.

R5 External Dosimetry

a. Inspection Scope

The inspectors reviewed selected data from the certificatee’s external dose monitoring program, reviewed technical basis documents, and discussed program implementation and results with cognizant personnel.

b. Observations and Findings

b.1 Portsmouth

External dose was monitored by means of TLD badges supplied by a vendor. The dosimeter contained 4 lithium-fluoride (LiF) elements, 3 lithium (^{7}Li) for beta and gamma dose measurements and one lithium (^{6}Li) for neutron measurements. The filtration over the elements was about 350 milligrams per square centimeter (mg/cm^2) for eye dose assessment, 1,000 mg/cm^2 for the deep dose, 17 mg/cm^2 for the shallow dose, 310 mg/cm^2 for the neutron and eye doses. The stated lower limit of detection for gamma dose measurement was 10 millirem (mrem). The dosimetry service that provided the dosimeters to the site and processed the dosimeters at the end of the badging period was accredited by National Voluntary Laboratory Accreditation Program (NVLAP).

The major sources of external radiation photons onsite were the uranium cylinders, with the highest dose rates coming from newly emptied cylinders. Spectra taken by the certificatee of the gamma radiation emitted by these empty uranium cylinders showed three prominent peaks: 185 keV from ^{235}U, and about 700 and 1,000 keV from protactinium (^{234m}Pa). Some material was left behind in the cylinders after the solid UF_6 was heated in the autoclave, liquefied, and then drawn off as a gas into the cascade. The remaining material consisted of some leftover uranium and uranium decay products and also transuranics that may have been in the feed material. Records of radiation surveys of newly emptied cylinders show contact dose rates on the order of 300 - 500 mrem/hr, with the dose rates dropping off to around 20 - 30 mrem/hr at 30 centimeters (cm). The dose rates decayed rapidly, reaching half these values in about 3 weeks.

Spectra from full cylinders showed a greater proportion of continuous distribution, believed to be partly scattered radiation from the main gamma rays and partly bremstrahlung from the absorption of high energy beta radiation emitted by uranium daughters. Because of a self-shielding effect, the dose rates from the full cylinders were much lower. Surveys by the inspectors showed gamma dose rates of the order of 2 - 3 mrem/hr at 30 cm, and similar dose rate levels from neutron radiation measured by the inspectors using rem-balls. According to a study conducted by the National Institute for Occupational Safety and Health (NIOSH), the neutrons arise from three sources: spontaneous uranium fission, induced uranium fission, and alpha-neutron (α,n) reactions with fluorine. The NIOSH study also suggested that the neutron energies were in the range of 200 - 400 keV, with the energy increasing with uranium enrichment, reaching approximately 500 keV at 2 percent ^{235}U and 750 keV at 5 percent. Thermal neutrons did not appear to be a dose contributor at the site.

The other potential sources of radiation exposure at the site were the beta, gamma and neutron calibration and check sources, of which there was a wide variety, uranium deposits in the cascade, as well as soft beta radiation at locations where ^{99}Tc may have concentrated. In-leakage of air into the cascade, which was operated at low pressure, causes a reaction with the UF_6 gas that leads to deposits of uranium on the inside surfaces of the cell walls. These deposits can build up and grow in mass over time, eventually forming a subcritical neutron pile and becoming a source of neutron exposure to personnel outside the cell. There were no estimates of the neutron doses that may result from such buildups, but the source locations were routinely monitored for size by the certificatee, and were removed if the source becomes too large. Another source of exposure was the ^{99}Tc, which was found in various parts of the system, but mainly in the purge traps at the upper end of the cascade. The principle hazard from ^{99}Tc is mainly as a skin contaminant because it tends to bind to the skin and is very difficult to decontaminate. Protective clothing was used in areas with a risk of ^{99}Tc exposure; however, high concentrations of ^{99}Tc may present a significant skin and eye external exposure hazard.

The dosimeter exchange period at the site was one year, except for certain workers with the potential for higher than site average external exposures, in which case the exchange period was quarterly. At the site, the only workers who fell into that category were those who handled the uranium cylinders, especially those who handled freshly emptied cylinders. The dosimetry records from 1990 to 1998 show that 3,000 - 3,600 persons were issued dosimetry each year (dosimetry records prior to March 3, 1997, were to be maintained in accordance with DOE requirements). The number of people with measurable doses had been steadily dropping, from about 1300 in 1990 to 190 in 1998. The collective dose for the site had also been steadily dropping, from about 46 person-rem in 1990 to 10 person-rem in 1998. The annual dose for the highest exposed individual had not shown the same declining trend, and ranged from 440 mrem in 1990, to a high of 800 mrem in 1994, and about 290 for the past 2 whole years. The number of workers that exceeded NRC's level requiring monitoring was 3 in 1993, 2 in both 1994 and 1995, and none for the other years between 1990 - 1998.

The inspectors noted that the site dosimeters had not been tested to verify that the vendor's algorithm for converting dosimeter light output during processing to dose equivalent was sufficiently accurate when used in the radiation environment at the site. The inspectors noted that quality assurance practices indicate that such a verification of dosimeter performance was appropriate. The certificatee stated that the vendor was NVLAP accredited, and as such, verification was unnecessary. However, ANSI Standard HPS N13.11-1993, "Personnel Dosimetry Performance - Criteria or Testing," stated that "This standard tests processor competence, not dosimetry system performance for all possible exposure conditions. . . . . . Processors should not calibrate their dosimetry systems with the test standard sources just to ensure that they pass performance tests, if the cost is poorer dosimetry performance in the workplace." The dosimeters used onsite were calibrated using the performance standard sources, but the performance onsite had not been verified for either photon or neutron radiations. Although it was most probable that the dosimeter performance will be found to be satisfactory, because most of the radiations onsite did not differ very significantly from those used in the testing standard, this should be verified. The dosimeter's response to beta radiation fields is even more dependent upon onsite conditions than the photon response, and the vendor's use of standard beta sources will probably not be adequate for accurate beta dosimetry for the beta fields onsite. This is especially the case because the site beta fields vary in energy with extremes from the high energy fields from ^{234m}Pa to the soft fields from ^{99}Tc. In addition, some form of quality control on the vendor should also be routinely undertaken. The subject of verifying vendor's algorithm for converting thermoluminescent (TL) output will be tracked as an **Inspection Follow-up Issue (IFI) 70-7002/99013-04** pending review and resolution by the certificatee.

c.1 Portsmouth Conclusions

The teams' review of the dosimetry program at PORTS concluded that it was adequate for most monitoring situations onsite. However, the ability of the dosimeter to accurately quantify doses at all site locations had not been verified, and there was no routine quality control program to ensure consistently high vendor performance. One **IFI** regarding verification of the algorithm for converting thermoluminescent output was identified.

b.2 Paducah

The same external dosimetry vendor, and the same badge design, were used at PGDP as described above at PORTS. Dosimetry was issued to any person who entered the restricted areas of the plant. Most personnel were on annual badging, but workers who had the potential to exceed 500 millirem (mrem) external dose in a year were on a quarterly badged cycle. About 3,000 - 4,000 workers were issued dosimetry each year.

A review of the 1998 external exposure record showed that 3,029 workers were monitored that year, and the collective dose for the site was 13.4 person-rem. Only about 200 of the

3,029 monitored workers showed measurable dosimeter readings. The highest deep dose equivalent was 382 mrem. About 25 percent of the collective dose was accumulated by workers in the C-400 Building, which was the decontamination and cylinder cleaning building, and the UF_6 handling shifts accumulated another 28 percent of the collective dose, with the health physics staff accumulating a little under 6 percent. These three groups thus accounted for over half of the site collective dose. The groups accumulating the highest collective doses also showed the highest individual doses.

The inspectors noted that, similar to PORTS, the site dosimeters had not been tested to verify that the vendor's algorithm for converting TL output to dose equivalent was sufficiently accurate when used in the radiation environment at the PGDP site. The subject of verifying vendor's algorithm for converting TL will be tracked as an **Inspection Follow-up Issue (IFI) 70-7001/99013-05** pending review and resolution by the certificatee.

c.2 Paducah Conclusions

The teams' review of the dosimetry program at PGDP concluded that it was adequate for most monitoring situations onsite. However, the ability of the dosimeter to accurately quantify doses at all site locations had not been verified, and there was no routine quality control program to ensure consistently high vendor performance. One **IFI** regarding verification of the algorithm for converting thermoluminescent output was identified.

R6 Internal Dosimetry

a. Inspection Scope

The inspectors reviewed selected data from the certificatee's internal dose monitoring program, reviewed technical basis documents, and discussed program implementation and results with cognizant personnel.

b. Observations and Findings

b.1 Portsmouth

Assessment of internal dose at the site was based on a urinalysis program. Monthly urine samples were collected from all workers with the potential for an intake of uranium exceeding 1 milligram per week (mg/wk). Workers with the potential for a uranium intake exceeding 0.3 mg/wk provided urine samples every 3 months. Notices were sent to all workers in the program reminding them to provide a urine sample by a specified date. Sealed plastic containers for collecting the samples were available to the workers at several locations onsite. Workers obtained the containers, provided the samples, and deposited the samples in one of several refrigerators provided for this purpose at a number of locations onsite. Health Physics staff collected those samples for analysis. Sample volumes collected were usually on the order of 60 -100 milliliters (mL). The samples were analyzed for uranium using mass spectroscopy, and workers with over 0.5 micrograms per liter (μg/L) uranium were re-sampled. The background uranium in urine level for the unexposed population near the PORTS site was 0.1 to 0.13 μg/L, and 0.5 μg/L represented the minimum level that could be distinguished reliably above background. Three quality assurance (QA) samples belonging to fictitious workers were submitted with each batch of routine samples. The QA samples were spiked with uranium of different enrichments at concentrations traceable to National Institutes of Standards and Technology (NIST).

The certificatee's internal dosimetry program was based on the concept of scaling used in many parts of the nuclear industry. Scaling was used when several radioactive materials always co-exist in roughly constant proportions, but it was relatively easy to measure one and very difficult or impossible to measure the others with an adequate sensitivity. Scaling was used at the site to estimate the intakes of small amounts of transuranic materials, which were difficult to measure with adequate sensitivity for routine monitoring purposes, based on an analysis of uranium in urine, which was easy to measure routinely.

The scaling ratios for the site were established in 1994 and were determined on the basis of a large number of contamination smears taken from various locations in the plant, as well as some air samples. The samples were analyzed for isotopic composition, and the ratios of uranium to other radionuclides present were determined for each sample. The analyses showed the presence of ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{237}Np, ^{228}Th, ^{230}Th, and ^{232}Th. (U uranium, Pu plutonium, Np neptunium, Th thorium). The data showed the following fractional activities for uranium, thorium and the transuranics:

TABLE 1.
Percentage of Total Activity in Samples Collected from Various Location

Location	U-234	U-235	U-238	Pu/Am	Np-237	Th-228	Th-230	Th-232
Cascade	78.1	3.1	18.7	0.014	0.014	0.014	0.031	0
Feed	81.4	3.3	15.3	0.01	0.01	0.003	0.029	0
X-705	89.6	2.9	7	0.047	0.035	0.1	0.31	0
X-710	91.9	2.7	4.7	0.072	0.18	0.035	0.39	0
Other	83	3.3	13.6	0.024	0.028	0.035	0.11	0

The data in Table 1 was based on a total of about 150 air samples and contamination smears. The certificatee used the sample data to calculate, for each sample, the ratio of the total Pu, Np, and Th to the total activity in the sample. The results are shown below in Table 2.

TABLE 2.
Percentage of Pu, Np, and Th to total activity in he samples.

Location	Average	Standard Deviation	Maximum
Cascade	0.23	0.34	1.42
Feed	0.19	0.23	0.78
X-705	1.0	1.2	7.9
X-710	0.84	2.1	12.2
Other	0.92	1.7	5.9

Based on the detailed data used to produce the above tables, the certificatee decided to use a value of 2 percent for transuranic isotopes in X-705 and X-710, and 0.5 percent for other areas of the plant. The justification for these values was that, although the maximum ratios were higher than the selected ratios, most of the ratios were below 2 percent for X-705 and X-710 (94 percent of the values), and below 0.5 percent for the other areas (86 percent of the values), and therefore the maxima were not representative of the environment to which the workers were exposed.

The certificatee assumed that all of the measured uranium in urine was ^{234}U. Because most of the uranium activity at the site is due to ^{234}U, and ^{234}U (ICRP Class D) was more radio-toxic on a per unit intake basis than the other uranium isotopes, that assumption was conservative. In addition, the certificatee assumed that all of the non-uranium activity was due to ^{230}Th. On a per unit intake basis, ^{230}Th (Class Y) is about as radio-toxic as ^{239}Pu (Class Y). However, ^{237}Np (Class W) is more radio-toxic than ^{230}Th and ^{239}Pu by a factor of about 3 for bone surface dose. Therefore, assuming that the Np activity was Th, as the certificatee did, was not conservative. However, there was much more Th than Np in the samples taken at the site, by factors ranging from about 3 to 11. Taking this into account, the assumption that all non-uranium activity was ^{230}Th was reasonable. The procedure the certificatee used to assign dose on the basis of urine analysis was to determine the concentration of uranium in the urine sample, assume that the activity was all ^{234}U and represented 98 percent of the total activity in the urine for workers in X-705 and X-710, and 99.5 percent for all other workers. The balance of the activity (2 percent and 0.5 percent, respectively) was assumed to be ^{230}Th. The time of intake was needed in order to calculate the dose based on a urine sample result. For known incidents that led to the intake, the time was generally accurately known. However, for routine urine samples that show an intake, the certificatee re-sampled all workers with positive urine results, and used the initial and re-sample urine concentrations to estimate the time of intake. A best-guess time was used initially, and was then adjusted until the data provided the best fit to the theoretical model. This method provided a reasonable estimate of time of intake and, although only two data points were a minimum for such model fitting, the uncertainty in such estimates was probably acceptable considering the uncertainties in applying the model itself.

A comparison of the 1993 survey data with data collected and analyzed in 1999 showed that the average ratio of non-uranium to total activity in the samples from all locations in the plant dropped from about 0.7 percent in 1993 to 0.06 percent in 1999. There was a corresponding drop in the cascade from 0.2 percent in 1993 to 0.03 percent in 1999, and for X-705 from 1.0 percent to 0.1 percent. There appeared to be a drop by an order of magnitude throughout the plant compared to 1993 values, and the 2 percent and 0.5 percent ratios currently in use were likely to be conservative for most intakes.

In calculating internal dose due to intakes of uranium, the certificatee did not use the standard ICRP 30 model for soluble uranium. Instead, a different model, the Fischer-modified, Wrenn-Lipsztein model was used. The certificatee stated that their uranium urine data did not fit the standard model very well, but was well described by the Wrenn-Lipsztein model. This model was also described in NUREG/CR-5566. The most significant difference between the standard and modified model was that the modified model predicted a more rapid initial uranium in urine excretion than the standard model, but more long-term bone retention.

A review of the certificatee's 1998 urine analysis result summary showed that 209 samples showed uranium activity above 1 µg/L, of which 185 were below 5 µg/L and all but one were below 40 µg/L.

c.1 Portsmouth Conclusions

The team concluded the certificatee's program for internal dose assessment at PORTS was adequate. However, the technical basis document describing that program and its connection with the air sampling program and other sources of isotopic information was not up to date.

b.2 Paducah

The internal dosimetry program at PGDP was similar to that at PORTS. About 9,000 urine samples were collected and analyzed each year, on a monthly collection cycle, and any worker with a potential for intake of radioactive material was placed in the program. Urine analyses were performed onsite using fluorimetry, in a procedure that was capable of measuring uranium in urine to a level of about 3 µg/L. There were about 500 to 600 workers in the program at the time of this inspection. Workers that provided samples showing uranium concentrations above 5 µg/L were re-sampled, and results above 20 µg/L were investigated to determine the cause of the intake. Samples above 40 µg/L were sent to Oak Ridge National Laboratories for isotopic analysis. A review of the urine analysis records for 1998 showed that about 40 samples were in the range of 5 - 19 µg/L, 5 were in the range of 20 - 39 µg/L, and 3 exceeded 40 µg/L. The certificatee did not perform any other bioassay measurement for the 3 workers that exceeded the highest action level of 40 µg/L. The inspectors noted that the certificatee had no procedural requirement for conducting fecal sample analysis, if high intakes occur, in which case fecal analysis may yield valuable additional early information for use in dose assessment.

Internal dose assessment based on urine analysis results was performed using the standard ICRP biokinetic models. The assumptions were made in these calculations that the intake consisted entirely of uranium and that it occurred 3 days prior to collection of the urine sample. The certificatee stated that available air sample data showed that there was very little TRU onsite and that the assumption of zero TRU was therefore warranted. The certificatee also stated that the 3 day period was used for time of intake because that was the time period beyond which any intake that was measurable by the site fluorimetry technique would be detected by the worker because of the irritant effect generated by any hydrogen fluoride (HF) contamination in the air (UF_6 when released in air, forms HF and UO_2F_2). In such cases, the worker would know the time of intake, and that would be used in the calculations.

The inspectors noted that the assumption of zero TRU onsite was difficult to support in view of available data. As discussed in the sections onsite characterization and air sampling above, characterization data from 1993 and 1994 showed the presence of ^{237}Np in most areas surveyed. Current comparable data was not available to update these surveys and, although it may be reasonable to assume that the 1994 levels of TRU had decreased in the intervening years because no fresh TRU had been introduced into the system for a very long time, this should be demonstrated rather than assumed. The available current data was in the form of a large amount of air sampling results obtained from the low volume area monitoring system, amounting to over 50,000 samples collected over a 3 year period. Most of these samples did not exceed the action level for isotopic analysis, and were therefore not analyzed for isotopic content. Some samples with low activity were composited and then sent for isotopic analysis. In total, about 27 air samples were analyzed for isotopic content, and many of those showed ^{237}Np to be present in varying fractions of activity.

The inspectors observed that the available data did not support the zero TRU assumption, and that the certificatee required a stronger basis for that assumption, if it was valid, especially since the SAR provided a value of 8 percent TRU as representative of site conditions. The subject of what percentage of TRU or similar activity to apply in the internal dose assessment program will be tracked as an **Inspection Follow-up Issue (IFI) 70-7001/99013-06** pending review and resolution by the certificatee.

The inspectors also noted that the argument for using 3 days post sampling as the time of intake was difficult to justify in many situations because it was conceivable, if not likely, that uranium contamination could exist without the accompanying irritant effect of HF at the higher concentrations. For example, if the source was the cascades or cylinders, or other systems containing UF_6, the products of UF_6 interaction in the air, HF and UO_2F_2, may not follow identical paths because of differences in density, and may not both reach the worker. In addition, the source of the contamination may be previously deposited contamination in the form of an oxide, the irritant vapors no longer being present. The inspectors noted that if assumptions were made that the intake occurred midway through the sampling cycle, as is often done in the nuclear industry, and the intake was assumed to include 2 percent ^{237}Np activity, as was suggested by available data, then the assessed internal dose for a given urine analysis result would increase by over an order of magnitude. It was therefore necessary for the certificatee to reassess these assumptions and provide firm justifications for using them. The inspectors noted that even with the more conservative assumptions made above, the assessed internal doses in most cases will still be only small fractions of the regulatory limit. The certificatee's reassessment of assignment of time to a given intake will be tracked as an **Inspection Follow-up Issue (IFI) 70-7001/99013-07** pending review and resolution by the certificatee.

c.2 <u>Paducah Conclusions</u>

The team concluded the certificatee's program for internal dose assessment at PGDP was adequate in most cases. However, some of the assumptions on which the program was based were not adequately supported by current data and, as a result, the internal dose assessments in some situations may not have been accurate if the assumptions were proven to be invalid. There was no adequate technical basis document that described the program and that described actions to be taken in case of an unexpected significant intake, such as whole body counting, fecal analysis, prophylaxis, or other therapy. The connection of the internal dose program with the air sampling program, and the use of the latter as a source of information for the former, was also not clearly documented. Two **IFIs**, one regarding the percentage of TRU or similar activity to apply in the internal dose assessment program and the other one regarding the assignment of time to a given intake were identified.

ENVIRONMENTAL AND EFFLUENT MONITORING

a. <u>Inspection Scope</u>

The inspectors reviewed the certificatee's programs for the assessment and control of releases of radioactive airborne emissions and liquid effluents, and evaluated the environmental and effluent monitoring programs for the assessment and characterization of radiological contaminants in the environs surrounding the sites. The primary program areas reviewed included: the environmental sampling program for ambient air, soil, sediment, vegetation and direct radiation; and the sampling and assessment of airborne emissions and liquid effluents through controlled release pathways.

The inspectors reviewed site specific data for the meteorological programs described in Section 5.1 of the SAR for each site. Meteorological data was required for the assessment of potential offsite radiological dose consequences from airborne emissions from the GDPs. Dose assessment calculations normally required meteorological input in the form of an annualized joint frequency distribution of wind speed, wind direction and atmospheric stability.

Environmental Monitoring

The inspectors reviewed the certificatee's programs for radiological environmental monitoring. These programs were described in Section 5.1 of the appropriate Safety Analysis Report. Radiological environmental monitoring programs (REMP) were conducted at the GDPs to assess the impact of the GDPs operations on the environment in the immediate vicinity of the plant sites. The REMP supplemented the radiological effluent monitoring program by verifying anticipated concentrations of radioactivity in the environment and related exposures to members of the public. Sample locations were selected based on meteorological data as well as other considerations, such as the locations of gardens, for example, in order to monitor exposure pathways important to human exposure.

Environmental samples that were collected and analyzed were compared against various predetermined action levels. Reviews of anomalous environmental sample results were triggered by action levels that were based on previous years of data and background data. A review of various sampling data identified a few cases where action levels were high enough to trigger a review. In each case the certificatee was able to demonstrate through discussion and documentation that the cause of the elevated value determined by USEC was reasonable and that there was no health and safety significance in each case.

R7 Ambient Air

b. Observations and Findings

b.1 Portsmouth

Fourteen permanent ambient air collection stations were being maintained. Three were located onsite, eight were located offsite around the reservation boundary, and three were located further offsite, which included one upwind of the prevailing site wind direction to collect background data. Each station contained a continuous low-volume air sampler with a membrane filter for collection of radioactive particulates. Most stations also contained a continuous high-volume air sampler with an 8 inch x 10 inch glass microfibre filter for collection of radioactive particulates. The high-volume particulate filters were collected weekly and the low-volume membrane filters were collected monthly. The filters were analyzed for gross alpha and gross beta-gamma.

The inspectors visited three of the air monitoring stations and observed a technician change out the filters. The monitoring stations were observed to be of high quality, were generally well maintained, and the technician was knowledgeable and proficient in changing the filters. It was noted that the low-volume sampler pump at Station A15 could not pull a vacuum, before or after the filter was changed. The technician indicated that it had been this way for some time and that a Maintenance Service Request had been written to get the problem fixed, but requests of this nature were typically a low priority maintenance item.

c.1 Portsmouth Conclusions

The ambient air sampling program was determined to be adequate to collect representative air samples for the assessment of radioactive particulate releases from the site.

b.2 Paducah

Five permanent ambient air collection stations were being maintained. Each station contained two continuous high-volume ambient air samplers, of which only one was operated at a time. The samplers used large filter papers to collect particulates from the air, and their use was switched back and forth weekly when the filters were collected. The filters were analyzed for gross alpha and gross beta. The inspectors visited four of the air monitoring stations and observed a technician change out the filters at two of the stations. The monitoring stations were observed to be of high quality, were well maintained, and the technician was knowledgeable and proficient in changing the filters.

c.2 Paducah Conclusions

The ambient air sampling program was determined to be adequate to collect representative air samples for the assessment of radioactive particulate releases from the site.

R8 Soil

b. Observations and Findings

b.1 Portsmouth

Soil samples from 46 locations surrounding the site were collected and analyzed on a semi-annual basis to monitor the environmental impact of the plant's emissions on the soil surrounding the site. Fifteen samples were collected at internal site locations, nine at onsite locations, six at offsite locations, twelve at remote offsite locations, and another four at 10 mile points from the site. Soil samples were analyzed for total uranium, gross alpha, gross beta and technetium. Action levels which would trigger further review were established based on previous years' data and the results of the background samples.

The inspectors visited six of the locations where soil samples were collected. Each of the locations were considered to be well suited for the collection of soil samples. All were in undisturbed areas where agricultural or other activities that could disturb the soil would not be expected to occur, and a minimal amount of gravel or rocks were present.

c.1 Portsmouth Conclusions

The Portsmouth soil sampling program was determined to be adequate to monitor the environmental impact of the plant's emissions on the soil surrounding the site.

b.2 Paducah

Soil samples from ten locations surrounding the site were collected and analyzed on an annual basis to monitor the environmental impact of the plant's emissions on the soil surrounding the site. Two of these locations (13 km south and 15 km west) were for background comparisons, while the others were at various directions from the site, including the prevailing downwind directions. Soil samples were analyzed for total uranium only. Action levels which would trigger further review were established based on previous years' data and the results of the background samples.

The inspectors visited three of the locations where soil samples were collected. Each of the locations was considered to be well suited for the collection of soil samples. All were in undisturbed areas where agricultural or other activities that could disturb the soil would not be expected to occur, and a minimal amount of gravel or rocks were present.

c.2 Paducah Conclusions

The soil sampling program was adequate to monitor the environmental impact of the PGDP plant's emissions on the soil surrounding the site.

R9 Sediment

b. Observations and Findings

b.1 Portsmouth

Sediment samples from 17 locations surrounding the site, of which four were at 10 mile points from the site, were collected and analyzed on a semi-annual basis to monitor the environmental impact of the plant's emissions on the soil surrounding the site. Sediment samples were analyzed for total uranium, gross alpha, gross beta and technetium. Action levels were established based on previous years data and the results of upstream sediment samples. The inspectors visited eight of the locations where sediment samples were collected. Each of the locations were observed to be well suited for the collection of sediment samples.

c.1 Portsmouth Conclusions

The sediment sampling program was adequate to monitor the environmental impact of the plant's operation to assess for the accumulation of radiological contaminants in receiving streams around the site.

b.2 Paducah

Sediment samples from six locations surrounding the site were collected and analyzed on an annual basis to assess for the accumulation of radiological contaminants in receiving streams around the site. Sediment samples were analyzed for total uranium, ^{235}U, ^{230}Th, ^{99}Tc, ^{237}Np and ^{239}Pu. Action levels were established based on previous years data and the results of upstream sediment samples. The inspectors visited three of the locations where sediment samples were collected. Each of the locations were considered to be well suited for the collection of sediment samples.

c.2 Paducah Conclusions

The sediment sampling program was adequate to monitor the environmental impact of the plant's operation to assess for the accumulation of radiological contaminants in receiving streams around the site.

R10 Vegetation/Produce

b. Observations and Findings

b.1 Portsmouth

Vegetation samples were collected semiannually from the same onsite, offsite and remote sampling locations where soil samples were collected. Vegetation samples consisting of wide-blade grass (forage for grazing animals) were collected and analyzed for total uranium and technetium. When available, produce samples were also collected within 16 kilometers from the site from local farmers and gardeners, and analyzed for uranium and technetium. Action levels which would trigger further review were established based on previous year's data and the results of the background samples.

The inspectors visited six of the locations where vegetation samples were collected. Each of the locations were considered to be well suited for the collection of vegetation samples.

c.1 Portsmouth Conclusions

The vegetation sampling program was determined to be adequate to monitor the environmental impact of the plant's emissions on the vegetation surrounding the site.

b.2 Paducah

To assess the impact of the plant's operation on the vegetation surrounding the site, various food crops were sampled on an annual basis. Crops collected were dependent on what was grown in the area, but were primarily corn, squash and tomatoes. Although samples were collected from seven locations (one was a background sample), the locations could vary from year to year depending on the prevailing wind directions and availability of crops. A review of the results of the analysis of vegetation samples from 1998 determined that the samples were analyzed for total uranium and ^{99}Tc as committed to in the Safety Analysis Report, as well as for ^{237}Np, ^{239}Pu, ^{234}U, ^{235}U, and ^{238}U.

c.2 Paducah Conclusions

The vegetation sampling program was adequate to monitor the environmental impact of the plant's emissions on the vegetation surrounding the site.

R11 Direct Radiation

b. Observations and Findings - Portsmouth/Paducah

There were 19 locations surrounding the PORTS facility where external gamma radiation was monitored. Monitoring was conducted using thermoluminescent detectors (TLDs) located at nine onsite locations, eight offsite locations, and two at locations distant from the site for background comparisons. The TLDs were collected and read quarterly. A review of direct radiation data for calendar year 1998 indicated all locations were well below the action levels.

There were at least 15 locations surrounding the PGDP facility where external gamma radiation was monitored. Monitoring was conducted using thermoluminescent detectors (TLDs) located at the site perimeter fence, reservation boundary, nearby residences and communities, as well as at locations distant from the site for background comparisons. The TLDs were collected and read quarterly. A review of direct radiation data for 1998 and the first quarter of 1999 indicated all locations were well below the action levels.

c. Portsmouth/Paducah Conclusions

The direct radiation monitoring program being conducted at both PORTS and PGDP was adequate.

R12 Meteorological Programs

b. Observations and Findings - Portsmouth/Paducah

At PORTS, an onsite meteorological tower measuring wind speed, wind direction and temperature at 10, 30 and 60 meters had been maintained. Data from this tower had been used as input for dose assessment calculations. Discussions with the certificatee indicated the instrumentation was being well maintained and calibrated as required. However, it was noted that atmospheric stability class had been determined by subtracting the temperature readings between two different elevations. Normal industry practice is to measure temperature differential directly with a set of matched thermocouples to meet standards for accuracy. ANSI/ANS-2.5-1984, Standard for Determining Meteorological Information at Nuclear Power Sites, specified an accuracy of ±0.5°C for temperature but ±0.15°C per 50 meters for temperature differential. Temperature sensor accuracy is usually insufficient to obtain an accuracy of ±0.15°C per 50 meters through the subtraction of two temperature readings. It was also noted that when data was missing, steps were taken to “fill in” the missing portions using interpolation or other assumptions. Since dose assessments normally require an annualized joint frequency distribution of data, it is not normal practice to fill in missing data unless it is necessary to achieve a specified level of data recovery. ANSI/ANS-2.5-1984 specifies a joint data recovery of ninety percent for wind speed, wind direction and atmospheric stability is sufficient. The inconsistencies with normal industry practices will be tracked as an **Inspection Follow-up Issue (IFI) 70-7002/99013-05** pending review and resolution by the certificatee.

At PGDP, an onsite meteorological tower measuring wind speed, wind direction and temperature at 10 and 60 meters had been maintained. However, the data logger which stored the hourly readings had been deteriorating since 1993, and ceased functioning completely in 1998. As a result, an acceptable level of recoverable onsite data had been unavailable since 1993. In lieu of this, the certificatee was using a 5 year composite of onsite data from the years 1988 through 1992. To verify that this data was representative of the 1993 to 1998 onsite data, annual wind roses of National Weather Service (NWS) data from the nearby Barkley Field airport were compared to the 1988-1992 5 year composite. The comparisons indicted that the annual NWS airport data for 1993 to 1998 correlated well with the 1988-1992 5 year composite data. Based on this finding, the certificatee used the 5 year composite for their 1993 through 1998 dose assessment calculations.

c. Portsmouth/Paducah Conclusions

The team concluded the meteorological programs at PORTS and PGDP were adequate. Some inconsistencies with industry practices were identified and one **IFI** was identified.

Effluent Monitoring

The inspectors reviewed the certificatee's programs for the control of effluent releases from the gaseous diffusion plants (GDP). This review included effluent sampling, sample analysis, data management, and data assessment to demonstrate compliance with NRC regulatory requirements. The effluent control programs were detailed in Section 5.1 of the Safety Analysis Report (SAR) for each facility. Section 5.1 included a discussion of effluent sampling, methods of effluent control, as well as methods of data assessment to demonstrate compliance with applicable regulations. Effluent sample analysis was described in Section 5.7, Analytical Support, of the SAR.

Liquid and airborne effluents from the GDPs were controlled through the use of established action levels or Baseline Effluent Quantities (BEQs). BEQs were established for discharge points from the GDPs, and the actual effluent releases were compared to the established BEQs in order to determine if action was necessary to reduce or stop a particular effluent release. The action levels and BEQs were detailed in Section 5.1 of the SARs. The BEQs were based on historical data from normal operating conditions at the GDPs and were reviewed on an annual basis.

In 1992 each GDP submitted to the Environmental Protection Agency (EPA) a site Compliance Plan for National Emission Standards for Hazardous Air Pollutants (NESHAPS) for Airborne Radionuclides. These plans detailed the manner in which each site would demonstrate compliance with NESHAPS with regard to airborne effluent monitoring, sample analysis, and offsite dose assessment. The Compliance Plan also discussed the design basis for the airborne effluent monitoring program. The Compliance Plans were subsequently approved by the EPA. Although the certificatee, USEC, leased and operated the enrichment facilities under NRC regulation, the facilities were subject to NESHAPS (40 CFR 61, subpart H) because the facilities were still owned by DOE. By implementing the Compliance Plan, which assessed the dose consequences of airborne radionuclide releases, the certificatee can demonstrate compliance with the requirements of the NRC regulations in 10 CFR 20 for radiation exposure to members of the public. The offsite dose calculations were performed by the Oak Ridge National Laboratory for the GDPs using the effluent data supplied by the GDPs and computer codes approved by EPA.

For liquid radioactive effluent released at the GDPs there was no credible exposure pathway leading to a measurable dose to members of the public. However, radioactive liquid effluent releases from the GDPs were monitored and controlled to ensure compliance with NRC regulations.

R13 Airborne Effluents

b. Observations and Findings

b.1 Portsmouth

At PORTS, thirteen emission sources were sampled on a continuous basis to quantify airborne radioactive effluent emissions. Six of these emissions points were also monitored on a continuous basis by flow through ionization chambers. As at PGDP, minor radioactive airborne release sources were not continuously sampled, but other approved methods were used to estimate releases from these sources.

The inspectors examined the airborne radioactive sampling systems for the Top Purge Cascade, Side Purge Cascade, and E-Jet (Purge Cascade) vents. The samplers consisted of a series of two alumina traps which collected the samples that had been drawn from the effluent stream by way of an isokinetic sampling probe. Both the vent flow and the sample flow were continuously monitored. The certificatee had evaluated the sampling systems with respect to the applicable ANSI Standard for effluent sampling. The inspectors noted that the samplers were located next to the vent lines in order to minimize sample line loss. The inspectors observed the weekly exchange of the alumina traps for these three sampling points, including the preparation of the alumina for the transfer to the laboratory, and the initial gamma spectrometry screening measurements of the alumina by the individual who performed the sample exchange, prior to submission of the samples to the laboratory. The weekly sample was analyzed or screened prior to submission to the laboratory so that an initial quick assessment could be made of the airborne radioactive emissions to determine if the effluent releases were within normally expected values prior to waiting several days for laboratory results. If the screening results were in excess of normally expected values, then the certificatee could begin to take action prior to receiving the laboratory results. The weekly samples were analyzed for total uranium, ^{235}U, and ^{99}Tc by the laboratory.

The inspectors reviewed the annual NESHAPS Report for 1997 and 1998 which summarized the airborne radionuclide emissions from PORTS. For the year 1998 the effective dose equivalent (EDE) to the maximally exposed individual offsite from USEC operations was 1.69 mrem. For the year 1997 the EDE to the maximally exposed individual offsite from USEC operations was 0.12 mrem. The 1997 EDE was typical of the offsite radiation exposure from previous years operations. The EDE of 1.69 mrem was due to a calculated "unplanned" release in 1998 which was caused by a fire which occurred in the Side Purge Cascade. The majority of the 1998 offsite exposure resulted from the radioactive material released as a result of this calculated unplanned release due to the fire. Routine airborne radioactive emissions would have produced an EDE of 0.12 mrem to the maximally exposed member of the public.

The inspectors observed the output of the flow through ionization chambers in the control room in the X-326 Building and discussed the use of the monitors with the operations individuals present in the control room. Those individuals stated that the ionization chambers were utilized primarily for monitoring operational events and noting any changes in operating systems.

c.1 Portsmouth Conclusions

Based on the above observations, reviews, and discussions, the team determined that the certificatee had in place an effective program for the control of airborne radioactive emissions from the PORTS facilities. The staff was knowledgeable and experienced in the area of effluent monitoring. The program functioned as described in Section 5.1 of the SAR.

b.2 Paducah

The dominant effluent pathway by which members of the public could receive measurable radiation exposure was via the airborne pathway. At PGDP the major airborne release source was sampled on a continuous basis. This source was the C-310 Purge Vent Stack. Minor airborne release sources were not sampled continuously, but other methods were used to estimate releases from these points, such as periodic sampling, or the use of estimation methods established in the NESHAPS Compliance Plan. The results from periodic sampling were used for subsequent emissions estimates. These minor release sources would not have the potential to exceed an offsite dose to the public of greater than 0.1 millirem per year.

The inspectors examined the C-310 Purge Vent Stack sampling system and noted that the sampling system consisted of a series of three caustic bubblers or scrubbers. The system also included a sampler flow totalizer and a vent stack flow totalizer. The certificatee had evaluated the installation of the vent stack sampling system with respect to the applicable ANSI Standard for stack sampling. The inspectors also noted that the certificatee had evaluated the potential sampling line loss for the sampling system. Additionally, the inspectors witnessed the daily exchange of the vent stack sampler, including the preparation of the sample for transfer to the laboratory for analysis as well as the actual transfer to the laboratory. The daily sample was analyzed for total uranium and gross beta. The gross beta was used as a screening analysis for ^{99}Tc. If the gross beta result was above normally expected values, then a ^{99}Tc analysis was performed on the sample. Portions of the daily sample were also composited into monthly and quarterly composite samples. The monthly composite samples were analyzed for ^{99}Tc, and the quarterly composite samples were analyzed for ^{230}Th, ^{237}Np, and ^{239}Pu.

The inspectors reviewed the annual NESHAPS Reports, which summarized the airborne radionuclide emissions from the GDP, for 1997 and 1998. For 1997 the effective dose equivalent (EDE) to the maximally exposed individual offsite was 0.0017 mrem, and for 1998 the EDE for the maximally exposed individual offsite was 0.011 mrem. The inspectors also reviewed the data from the last periodic sampling of emission points which were sampled on a periodic (every 5 years) basis.

Discussions with the certificatee indicated that mechanisms were in place to identify any changes to facility systems which would add or effect effluent emission points. The certificatee reviewed and evaluated effluent emission points on an annual basis.

c.2 Paducah Conclusions

Based on observations, reviews, and discussions, the team determined that the certificatee had in place an effective airborne radioactive effluent control program at PGDP. The certificatee's PGDP staff were knowledgeable and experienced. The airborne radioactive effluent control program functioned as outlined in Section 5.1 of the SAR. Offsite doses to members of the public were a small fraction of the regulatory limits.

R14 Liquid Effluents

b. Observations and Findings

b.1 Portsmouth

Effluent water generated at the site included cooling tower blowdown water, once through cooling water, sewage treatment plant effluent, rain runoff, and process wastewater. Wastewater generated at the site came from decontamination and cleaning activities. This wastewater was processed in the decontamination and recovery facility (X-705, Decontamination Building and X-700, Cleaning Building) to remove radionuclides prior to being discharged to the onsite Sewage Treatment Plant. The Sewage Treatment Plant discharged to Outfall W003, which was one of eight USEC-leased water outfalls that were being monitored for radioactive effluents. W003 discharged directly to the Scioto River.

Wastewater generated in the decontamination and recovery facility was processed through a biodenitrification facility in Building X-700. However, before the waste stream could be transferred for biodenitrification, a sample was taken and it was verified that the technetium concentrations were less than 0.0002 grams/Liter (g/L) (0.2 ppm) and uranium content was less than or equal to 0.005 g/L (5.0 ppm). By ensuring that these criteria were met, eventual releases from the Sewage Treatment Plant would meet concentration guidelines for releases to uncontrolled areas.

A review of the analytical results of the uranium and technetium data from the waste stream prior to a transfer to the biodenitrification facility indicated the radiological levels in the waste stream were within the levels required. Additionally, the certificatee obtained the liquid radioactive effluent sample analysis results from the laboratory data base and calculated a waterborne dose to a hypothetical member of the public based on the drinking water and fish consumption pathways to demonstrate compliance with NRC requirements for radiation exposure to members of the public. The calculated dose was projected for the entire year. The calculated maximum 50-year committed effective dose equivalent (CEDE) for 1999, based on the sample data to date was 0.0022 mrem.

There were eight locations where USEC conducted effluent sampling of water that originated within the site. These locations were sampled for radiological contamination. Sampling for the presence of other non-radiological analytes based on the requirements of USEC's NPDES Permit issued by the Ohio EPA was also conducted at these outfalls. Because many of these outfalls did not maintain a continuous flow, the sampling included either continuous time proportional composite samplers or grab samples when flow was present at the outfall. The outfalls continuously sampled were sampled such that a weekly composite sample was collected. These samples were analyzed for gross alpha, gross beta, total U, and ^{99}Tc. USEC also obtained grab samples from Little Beaver Creek, Big Beaver Creek, Big Run Creek, and the Scioto River as part of the site REMP. These surface waters were sampled both upstream and downstream of USEC discharges into these surface waters. The Scioto River and one downstream location in Little Beaver Creek were sampled weekly. All other locations were sampled monthly. These samples were also analyzed for gross alpha, gross beta, total U, and ^{99}Tc.

The inspectors visited three of the outfalls and observed a technician collecting weekly composite samples at two of the locations. (The third location had no flow.) At the visited locations the sampling lines appeared to be adequately located for collection of a representative sample of the effluent from the outfalls. The sampling technician appeared to be knowledgeable and proficient. At the first observed sampling location (W001), the sample pump was clogged with algae. The technician identified the problem, cleared the pump intake, and checked the flow of the pump. Also at this site it was observed that the sample collection container contained less than the expected volume of water. It appeared that the sampler had not been collecting its expected 50 milliliters (ml) every hour in spite of the fact that the pump seemed to be running acceptably. The technician indicated that although he visited the site daily, it was not part of his routine to open the refrigerator (the samples were collected in a refrigerator) and inspect the cubitainer to verify that if was filling as expected. The technician resolved the low flow problem.

The inspectors reviewed USEC's data from the collection, sampling, and release of liquid effluents. The data indicated that the releases were being carried out as described in the SAR. The procedures and their implementation were determined to be adequate to prevent liquid radioactive effluent releases in excess of regulatory limits.

c.1 Portsmouth Conclusions

Overall the effluent water sampling program at PORTS was determined to be of high quality and adequate to collect representative water samples for the analysis of radioactive releases from the site and for the determination of doses to members of the public.

b.2 Paducah

Wastewater generated at the site included recirculating water (RCW) blowdown, nonprocess wastewater (e.g., once-through cooling water from pumps and air conditioners, drinking fountain drains, safety shower drains), process wastewater (e.g., decontamination and cleaning solutions, contaminated liquid wastes from laboratories), sanitary wastewater, and runoff. The RCW blowdown and nonprocess wastewaters were discharged directly to outfalls. Process wastewater was collected, treated and sampled prior to release to ensure compliance with 10 CFR Part 20 release limits for effluent releases. Where possible, treated process wastewater was reused onsite. Sanitary wastewater was treated at the Sewage Treatment Facility and then discharged through an outfall.

The PGDP maintains doses to the public from liquid releases within the regulatory limits by controlling the quantities of radiological materials in their releases. All processed waste streams that were collected and sampled prior to release were sampled for the presence of ^{234}U, ^{235}U, ^{238}U, ^{99}Tc, ^{237}Np, and ^{239}Pu. Using the sum-of-fractions rule and the release criteria from 10 CFR Part 20, Appendix B, Table 2, Column 2, if the sum of the fractions was less than one, then the sampled effluent could be released. If the sum of the fractions was greater than ten, then the effluent had to be treated and the radionuclides removed prior to release. If the sum of the fractions was greater than or equal to one and less than or equal to ten, the effluent could still be released, but only if there was a sufficient flow of water through the outfall where the release was to occur. This was determined by factoring in the volume of water that would exist from the normal flow rate at the outfall during dry weather. If the sum of the fractions was recalculated with this additional flow to show a result of less than one, then the discharge would be allowed. However, this discharge could only occur at a continuously sampled outfall (i.e., an outfall that had a continuous flow of water through it).

There were ten locations (outfalls) where USEC conducted effluent sampling of water that originated within the site security area. These locations were sampled for the presence of radiological contamination. Sampling for the presence of other non-radiological analytes based on the requirements of USEC's Kentucky Pollutant Discharge Elimination System (KPDES) permit was also conducted at these outfalls. Because many of the outfalls were not continuously flowing, but dependent upon runoff from precipitation for flow, the sampling included both continuous flow proportional samples as well as grab samples. Collection frequencies included the collection of weekly continuous flow samples at three outfalls, with monthly and quarterly composite samples made from the weekly samples, and monthly and quarterly grab samples from seven of the outfalls following the occurrence of adequate rainfall to allow collection. All outfalls were sampled at least quarterly. Quarterly samples were analyzed for ^{237}Np, ^{239}Pu, ^{230}Th, dissolved alpha, dissolved beta, suspended alpha and suspended beta. In addition, depending on the collection frequency for each outfall (weekly, monthly or quarterly), samples were also analyzed for total uranium, percent ^{235}U, and ^{99}Tc. USEC also sampled the receiving streams, Big Bayou and Little Bayou Creeks as part of their Radiological Environmental Monitoring Program (REMP). Big Bayou Creek was sampled both upstream and downstream of USEC discharges into these waters. Little Bayou Creek was sampled at only one location downstream of any discharges. All of the samples were grab samples and were taken monthly. The samples were analyzed for total U, gross alpha, gross beta, ^{99}Tc, ^{239}Pu, and ^{237}Np.

The inspectors visited seven of the outfall locations and observed two technicians collecting a weekly sample at two of the locations. All locations visited were observed to be adequately located for the collection of effluent water samples. Each site had a weir installed to improve the quality of the samples collected, the sampling equipment was of a high quality and well maintained, and calibration dates observed on equipment were current. The technicians were knowledgeable and proficient and no problems were noted with the sampling techniques. At one site the technician did not think the sample collection pump sounded right, so she took it apart, replaced the inner tubing, and rechecked the suction pressure.

The inspectors reviewed USEC's data from the collection, sampling and release of a number of liquid effluents. In all cases examined data indicated that releases were being carried out as specified by the SAR and procedures. The procedures and the observed implementations were noted to be acceptable to prevent radiological effluent releases in excess of regulatory limits.

c.2 Paducah Conclusions

Overall the effluent water sampling program at PGDP was determined to be of high quality and adequate to collect representative water samples for the analysis of radioactive releases from the site.

METHODOLOGY FOR MEASURING RADIONUCLIDES

R15 Analytical Support

a. Inspection Scope

The inspectors reviewed site specific procedures, observed laboratory activities, and conducted interviews of cognizant personnel to evaluate the methodology for measuring radionuclides. The radiochemistry laboratories that provided analytical support to the radioactive effluent control program and the radiological environmental monitoring program were so similar in operations, facilities, equipment, and procedures that the laboratories will be discussed together. Any significant differences are noted. The laboratory programs were described in Section 5.7 of the SAR for each site.

b. Observations and Findings - Portsmouth/Paducah

The inspectors toured selected portions of the laboratories and followed the delivery of the stack and vent effluent samples (from the C-310 stack at PGDP and the purge cascade vents at PORTS, respectively) and observed portions of the sample analyses. Samples were logged into the laboratory sample tracking systems after receipt which provided for sample chain of custody and accountability throughout the analytical process. The samples were analyzed according to approved laboratory procedures, and the results were entered into laboratory data bases. The sample results in the data bases were available for viewing by the individuals responsible for the radioactive effluent control and radiological environmental monitoring programs. The inspectors also discussed sample preparation and analysis techniques; instrument calibration techniques; the laboratory data bases; methods of data reporting, including propagation of uncertainties; and sample chain of custody with the laboratory staff.

The laboratories had in place laboratory QA/QC programs which included both an intralaboratory QC program and an interlaboratory QC program. The intralaboratory QC program consisted of the use of blank samples, duplicate samples, matrix spikes, sample spikes, the use of tracers, and instrument control checks. The interlaboratory QC program consisted of the analysis of performance evaluation samples supplied by outside laboratories such as the Department of Energy's (DOE) Environmental Measurements Laboratory (QAP Program) and DOE's Radiological and Environmental Sciences Laboratory (MAPEP Program). Control charts were used by the laboratory staff to plot and trend the results of QC analyses in order to provide control of the analytical process. The inspector's review of laboratory QC data for 1998 and 1999 to date indicated good performance by the laboratories.

The laboratories possessed state of the art analytical instrumentation, including alpha spectrometry systems, gamma spectrometry systems, gas flow proportional counters, liquid scintillation counters, and mass spectrometry systems. Instruments were calibrated with National Institute of Standards and Technology (NIST) traceable standards. Laboratory supplies, chemicals and reagents, along with other apparatus such as balances, centrifuges, glassware, and fume hoods were in adequate supply.

c. Portsmouth/Paducah Conclusions

Based on the above observations, reviews, and discussions, the team concluded that the certificatee's laboratory staff were knowledgeable and experienced. The laboratories were providing the required level of support to the radioactive effluent control and radiological environmental monitoring programs with respect to the required analytical detection levels for the types of samples submitted to the laboratory. Both laboratories had implemented QA/QC programs to ensure their continual adequate performance. The laboratories produced credible, defensible analytical data.

R16 Data Management

a. Inspection Scope

The inspector's reviewed data, observed handling of information, and discussed with cognizant personnel the data management system for the PORTS and PGDP sites.

b. Observations and Findings

b.1 Portsmouth

The inspectors reviewed the handling and storage of radiological analyses results which was comprised of a laboratory information system and an environmental data base. Analytical results were placed into the laboratory data base and were available for access or review by the individual responsible for the effluent and environmental monitoring programs. The data was electronically transferred daily to data bases and spread sheets used for evaluation and control of radioactive effluents from the facility, and the evaluation of data generated by the REMP.

Data were trended and evaluated against BEQs and background sampling points. Field data were also reviewed. If any limits were exceeded or inconsistencies identified, an additional review was conducted in order to resolve the inconsistency or identify the reason for a limit being exceeded.

c.1 Portsmouth Conclusions

The data management program at PORTS was determined to be well thought out and effectively implemented.

b.2 Paducah

The inspectors reviewed the handling and storage of radiological analyses results which was comprised of a laboratory information system and an environmental data base. Requests for radiological analysis were first placed into the laboratory information system. With the exception of the results of the C-310 stack data, the analytical results from these sample analyses requests were entered into the laboratory information system. These data were then electronically transferred daily to the environmental data base where the data underwent a quality assurance check and data verification. Data was compared against historical and statistical limits with anomalies automatically flagged. Data were manually checked to verify they were acceptable before they were downloaded for permanent storage. Because the C-310 stack sample data was used for process control purposes, these results were provided to the plant cascade coordinator in lieu of transferring the data to the environmental data base.

Data were trended and evaluated against BEQs and background sampling points. Field data, including log books and calculations were also reviewed. If any limits are exceeded or inconsistencies identified, an additional review was conducted which in some cases included the re-analysis of a sample. When a sample was re-analyzed, both analytical results were maintained in the data base.

c.2 Paducah Conclusions

The data management programs at PGDP was determined to be well thought out and effectively implemented.

V. Management Meeting

X Exit Meeting Summary

The inspectors presented the inspection results to members of the plant staff and management on October 27 and 28, 1999, at the Paducah and Portsmouth areas respectively. The plant staff acknowledged the findings presented. At the conclusion of each of the onsite inspection periods, the inspectors asked the plant staff whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

United States Enrichment Corporation

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M. Weber, Deputy Director, Fuel Cycle and Safeguards, NMSS*/**
R. Pierson, Chief, Special Projects Branch, NMSS*/**

*Denotes those present at the exit meeting October 27, 1999
** Denotes those present at the exit meeting October 28, 1999

Other members of the plant staff were also contacted during the inspection period.

INSPECTION PROCEDURES USED

IP 83822: Radiation Protection
IP 88035: Radioactive Waste Management

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

PADUCAH

70-7001/99013-01	IFI	Section R1.b.2 - Assess the need for small pockets of contaminated areas to be cleaned up.
70-7001/99013-02	IFI	Section R2.b.2 - Using SAR criteria for radiological posting and boundary control.
70-7001/99013-03	IFI	Section R3.b.2 - Review of current training material for accurate and site-specific information.
70-7001/99013-04	IFI	Section R4.b.2 - Assess the need to review the collection efficiency basis on high volume air samplers.
70-7001/99013-05	IFI	Section R5.b.2 - Verify vendor's algorithm for converting TL in the external dosimetry program.
70-7001/99013-06	IFI	Section R6.b.2 - Assess/correct the percentage of TRU or similar activity applied to the internal dose assessment program.
70-7001/99013-07	IFI	Section R6.b.2 - Assessment of the assignment of intake time to for a specific intake when the pathway was unknown.

PORTSMOUTH

70-7002/99013-01	IFI	Section R1.b.1 - Characterization and cleanup of the southeast corner of X-326.
70-7002/99013-02	IFI	Section R3.b.1 - Review and development of accurate and complete training material for radiation workers.
70-7002/99013-03	IFI	Section R4.b.1 - Assess the need to revise technical basis documents and procedures.
70-7002/99013-04	IFI	Section R5.b.1 - Verify vendor's algorithm for converting TL in the external dosimetry program.
70-7002/99013-05	IFI	Section R12.b - Assess/resolve inconsistencies with industry practices regarding meteorological temperature corrections.

LIST OF ACRONYMS USED

ALARA	As Low as Reasonably Achievable
Am	americium
ANS	American National Standards
ANSI	American National Standards Institute
BEQ	Baseline Effluent Quantities
BZA	Breathing Zone Samplers
CAM	Continuous Air Monitor
CCZ	Contaminated Control Zones
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
Ci	Curie
mCi	milli-curie, or $1x10^{-3}$ Curie
μCi	micro-curie, or $1x10^{-6}$ Curie
μCi/ml	micro-curie per milliliter
pCi	pico-curie or $1x10^{-12}$ Curie
cm^2	square centimeters
DAC	Derived Air Concentration
DMSA	DOE Material Storage Areas
DOE	Department of Energy
DOT	Department Of Transportation
dpm	disintegration per minute
EDE	Effective Dose Equivalent
EPA	Environmental Protection Agency
ft^3	Cubic-feet
GDP	Gaseous Diffusion Plant
GET	General Employee Training
HEPA	High Efficiency Particulate Air
HF	Hydrogen Fluoride
HP	Health Physics
HPT	Health Physics Technician
ICRP	International Committee on Radiation Protection
IDLH	Immediately Dangerous to Life and Health
IFI	Inspection Follow-up Issue (these issues will be reviewed further by Region III to assess compliance with commitments and/or requirements)
keV	Kilo electron-Volt
km	Kilometer
KPDES	Kentucky Pollutant Discharge Elimination System
Li	Lithium
LiF	Lithium Fluoride
L/min	Liters per minute
MAPEP	DOE's Radiological and Environmental Sciences Laboratory
MDA	Minimum Detectable Activity
MeV	Mega electron-Volt
mg	milligrams
mg/cm^2	milligrams per square centimeter
mrem	millirem or $1x10^{-3}$ rem
ml	milliliter
m^3	Meters Cubed or cubic meters
µg/L	micrograms per liter
NESHAPS	National Emission Standards for Hazardous Air Pollutants

NIOSH	National Institute for Occupational Safety and Health
NIST	National Institutes of Standards and Technology
Np	neptunium
NPDES	National Pollution Discharge and Emissions Standards
NRC	Nuclear Regulatory Commission
NRRPT	National Registry of Radiation Protection Technicians
NVLAP	National Voluntary Laboratory Accreditation Program
NWS	National Weather Service
ORNL	Oak Ridge National Laboratory
ORISE	Oak Ridge Institute for Science and Education
Pa	protactinium
PGDP	Paducah Gaseous Diffusion Plant
PIPS	Passivated Implanted Planar Silicon
PORTS	Portsmouth Gaseous Diffusion Plant
ppm	parts per million
PSS	Plant Shift Supervisor
Pu	plutonium
QA	Quality Assurance
QAP	Quality Assurance Program
QC	Quality Control
RCW	Recirculating Cooling Water
REMP	Radiological Environmental Monitoring Program
RPM	Radiation Protection Manager
RU	Recycled Uranium
RWP	Radiological Work Permit
SAR	Safety Analysis Report
Tc	technetium
Th	thorium
TL	thermoluminescent
TLD	Thermoluminescent Detector
TRU	Transuranic Nuclides, i.e. atomic number > 92
U	uranium
UF_6	Uranium Hexafluoride
USEC	United States Enrichment Corporation

LIST OF ELEMENTS USED

Throughout this report, basic elements are expressed in a standard engineering notation. The expressions used identify the element by its chemical symbol and the specific isotope by use of standard notations. For example, uranium (U) has an atomic number (i.e. number of protons) of 92, and uranium exists in several different isotopic forms denoted by the atomic mass (i.e., number of protons + number of neutrons). The most common isotope of uranium has 146 neutrons; therefore, its atomic mass = 238 (146 neutrons + 92 protons).

The general notation used in this report indicates the atomic mass number first followed by the chemical symbol for the specific element. Since the chemical symbol is provided, the atomic number does not normally appear.

ISOTOPE NOTATION	ELEMENT	ATOMIC NUMBER	ATOMIC MASS
^{6}Li	lithium	3	6
^{7}Li	lithium	3	7
^{99}Tc	technetium	48	99
^{228}Th	thorium	90	228
^{230}Th	thorium	90	230
^{232}Th	thorium	90	232
^{234m}Pa	protactinium	91	234
^{234}U	uranium	92	234
^{235}U	uranium	92	235
^{236}U	uranium	92	236
^{238}U	uranium	92	238
^{237}Np	neptunium*	93	237
^{238}Pu	plutonium*	94	238
^{239}Pu	plutonium	94	239
^{240}Pu	plutonium	94	240
^{241}Am	americium*	95	241

* Elements with atomic number greater than 92 are referred to as the transuranics.